Appendix D
Human Health Impacts from Facility Accidents

APPENDIX D
HUMAN HEALTH IMPACTS FROM FACILITY ACCIDENTS

D.1 Impact Assessment Methods for Facility Accidents

D.1.1 Introduction

This appendix presents the methodology and assumptions used for estimating potential impacts and risks associated with both radiological and hazardous material releases, due to postulated accidents, at the facilities being considered for the Versatile Test Reactor (VTR), VTR fuel production facilities, VTR post-irradiation examination facilities, VTR spent fuel treatment facilities, and VTR spent fuel storage facilities. Accidents involving transuranic (TRU) waste generated during operation of the VTR and associated facilities are also considered. Considerations for accident selection are presented in Section D.2. Selection of postulated accidents for the various alternatives and options and determination of the associated radiological source terms are presented in Section D.3. Radiological impacts from the accidents are presented in Section D.4. Assessment of hazardous material releases is discussed in Section D.5. Information regarding the impacts of normal operations, along with background information on the health impacts from exposure to ionizing radiation and exposure to hazardous materials, is provided in Appendix C.

D.1.1.1 Consequences and Risks

Metrics commonly used in environmental impact statements (EISs) to present the potential impacts of accidents are consequences and risks. The consequences are the potential impacts that would result if the accident were to occur. Accident consequences may be presented as impacts on individuals or a specified population (e.g., residents within 50 miles of an accident) and in terms of dose (e.g., rem or person-rem) or health effects (e.g., latent cancer fatalities [LCFs]). Risk is defined as the product of the consequences and estimated frequency of a given accident. The accident frequency is the number of times the accident is expected to occur over a given time (e.g., per year).

Accident consequences are determined using the MELCOR Accident Consequence Code System, Generation 2] (MACCS2) computer program/code (NRC 1990, 1998). A number of specific types of risk can be directly calculated from the output of the MACCS2 computer program. The risk to a noninvolved worker or to a maximally exposed member of the public (MEI) can be calculated. The MACCS2 computer code yields a dose to the noninvolved worker and the MEI. Using the risk factor of 0.0006 LCFs per rem (DOE 2003), the consequence in terms of the likelihood of an LCF can be calculated. The risk to this hypothetical individual is calculated by multiplying the consequence in terms of an LCF by the estimated accident frequency. For example, if an accident has an estimated frequency of 0.001 per year and the dose from the accident is 1 rem, the risk is $0.001 \times (1 \times 0.0006) = 6 \times 10^{-7}$ LCFs per year.

Calculation of the population risk is also possible. Population risk, which is the product of the total consequences experienced by the population and accident frequency, is a measure of the expected number of LCFs experienced by the population as a whole over the course of a year.[1] For example, if an accident has a frequency of 0.001 per year and the consequence of the accident is 5 LCFs in the population, then the population risk is $0.001 \times 5 = 0.005$ LCFs per year.

[1] Population distribution data for each facility considered in this VTR EIS is presented in Appendix C.

D.1.1.2 Uncertainties and Conservatism

The analyses of accidents are based on calculations relevant to hypothetical sequences of events and models of their effects. The models provide estimates of the frequencies, source terms, pathways for dispersion, exposures, and effects on human health and the environment that are as realistic as possible within the scope of the analysis. In many cases, minimal experience with the postulated accidents leads to uncertainty in the calculation of their consequences and frequencies. This fact has prompted the use of models or input values that yield conservative estimates of consequence and frequency. All alternatives have been evaluated using uniform methods and data, allowing for a fair comparison of all alternatives (DOE 2004b).

One method for evaluating alternatives involves comparing individual and population risks that can be calculated from the information in this VTR EIS. The equations for such calculations involve accident frequency and accident consequences, parameters that are subject to uncertainty. The uncertainty in estimates of the frequency of events can vary over several orders of magnitude. Similarly, consequence calculations depend on inputs that can vary by several orders of magnitude. The generally accepted practice is to report accident frequencies qualitatively, in terms of broad frequency bins, as opposed to numerically and to report consequences as numeric values from the consequence calculations. Therefore, in this VTR EIS, the consequence metrics have been preserved as the primary accident analysis results. Likewise, accident frequencies have been identified qualitatively, to provide a perspective on risk that does not imply an unjustified level of accuracy.

D.1.2 Safety Strategy

The VTR is being designed with the concept of ensuring safety throughout the proposed operating regions as well as being resilient to potential accidents or upsets whether they are caused by internal hazards (such as human errors, equipment failures, or fires) or external hazards (such as seismic events, vehicle impacts, or wind loading). Consistent with U.S. Department of Energy (DOE) guidance for safety in design, the VTR design focus is on providing capabilities that reduce or eliminate hazards, a bias towards preventive, as opposed to mitigative, design features. The design also demonstrates a preference for passive systems over active systems. This general approach creates a design which is reliable, resilient to upset, and has low potential consequences of accidents.

Safe operation of the VTR depends on reliable systems designed to ensure the key reactor safety functions are achieved. These key safety functions can be summarized as (1) reactivity control, (2) fission and decay heat removal, (3) protection of engineered fission product boundaries, and (4) shielding. The first three of these safety functions are generally relevant for safe VTR operations, both within the reactor and outside the reactor. Given the anticipated distances between the VTR candidate sites and the public boundaries, the shielding function for VTR facilities and activities is primarily only of concern for involved workers. However, shielding is also a potential concern for noninvolved workers and the public during irradiated material transports.

For VTR fuel production facilities, spent fuel storage and treatment facilities, and post-irradiation examination facilities, including the hot cell and glovebox facilities like those evaluated in this EIS, the general safety strategy requires the following:

- Confinement of fuel and fission product materials at all times that prevents the materials from reaching the environment.
- Minimization of energy sources that are large enough to disperse the plutonium and fission products and threaten confinement.

This basic strategy means that operational accidents, including spills, impacts, fires, and operator errors, never have sufficient energy available to challenge the confinement. The final layer of confinement is the system of barriers and multiple stages of high-efficiency particulate air (HEPA) filters that limit the amount of material that could be released to the environment.

The operational events that present the greatest threats to confinement are large-scale internal fires that, if they were to occur, could present heat and smoke loads that threaten the building's HEPA filter systems. For modern plutonium facilities,[2] the safety strategy is (1) to prevent large internal fires by limiting energy sources, such as flammable gases and other combustible materials, to the point that a wide-scale, propagating fire is not physically possible and (2) to defeat smaller internal fires with fire-suppression systems.

Modern hot cell and glovebox facilities for plutonium and fission product operations are designed and operated such that the estimated frequency of any large fire within the facility would fall into the extremely unlikely category and would require multiple violations of safety procedures to introduce sufficient flammable materials into the facility to support such a fire. Any postulated large-scale fire in a modern plutonium facility that would be expected to result in severe consequences if it occurred would be categorized as a "beyond-design-basis" event and would fall into the "beyond extremely unlikely" category.

Earthquakes present the greatest design challenges for these facilities due to the requirement to prevent substantial releases of radioactive materials to the environment during and after an earthquake. For safety analysis purposes, occurrence of a substantial release of radioactive material within the facility is often assumed in response to a postulated earthquake that exceeds the design loading levels of the facility equipment, enclosures, and building structure and confinement. This assumption allows designers and safety analysts to determine that design features are adequate to ensure confinement of the radioactive material at risk (MAR) for any seismic event up to and through the design-basis earthquake.

D.1.3 Selection of Accidents and Determination of Radiological Source Terms

Potential accident scenarios have been identified for the Idaho National Laboratory (INL) Materials and Fuels Complex (MFC), the Oak Ridge National Laboratory (ORNL) and the Savannah River Site (SRS) facilities. Alternatives for siting the VTR include locations at the INL MFC and ORNL. Options for siting the VTR fuel production facility include locations at the INL MFC and SRS. Alternatives for siting the post-irradiation examination facility and the spent fuel treatment and storage pad include locations at the INL MFC and ORNL.

The analysis of accidents is based on calculations relevant to hypothetical sequences of events and models of their effects. The analysis in this appendix includes accident scenarios that address duration of release, elevation of release, MAR, source terms (quantities of radioactive and hazardous materials released to the environment), and consequences. The models provide estimates of the frequencies, source terms, pathways for dispersion, exposures, and effects on human health and the environment that are as realistic as possible within the scope of the analysis. In many cases, minimal experience with the postulated accidents leads to uncertainty in the calculation of their consequences and frequencies. This fact has prompted the use of models or input values that yield conservative estimates of consequence and frequency. However, given the preliminary nature of the designs under consideration, quantitatively assessing the frequency of occurrence of the events addressed is not possible. Consequently, the frequency of occurrence is qualitatively assessed, and the event frequency is assigned to a bin.

In this analysis, four frequency bins/categories are defined. The frequency bins are selected based on DOE guidance for safety analyses and National Environmental Policy Act (NEPA) documents for facilities with similar operations. Accident frequencies are grouped into the bins of "anticipated," "unlikely,"

[2] Because the VTR fuel would be 20 percent plutonium, the facilities for feedstock preparation, fuel fabrication, and spent fuel treatment would be plutonium facilities.

"extremely unlikely," and "beyond extremely unlikely," with estimated frequencies of greater than 1×10^{-2}, 1×10^{-2} to 1×10^{-4}, 1×10^{-4} to 1×10^{-6}, and less than 1×10^{-6} per year, respectively. In this EIS, the frequency estimate includes both the initiating event and conditional events/conditions leading to the release. For example, an aircraft crash includes not only the frequency of an aircraft impacting a facility, but also the probability of the containment being breached and system damage resulting in core damage. The accident analysis considers accident scenarios that represent the spectrum of reasonably foreseeable accidents, including low-frequency/high-consequence accidents and higher frequency/(usually) lower consequence accidents. Typically, accidents with a frequency of less than 10^{-7} per year are not considered reasonably foreseeable and do not need to be examined. However, because of the effectiveness of advanced reactor safety systems, low-frequency/high-consequence accidents for the VTR typically have a frequency of less than 10^{-7} per year. Consequently, accidents with a frequency of less than 10^{-7} per year are considered in this EIS in order to provide insight into accidents with greater impacts (DOE 2008). All alternatives have been evaluated using uniform methods and data, allowing for a fair comparison of all alternatives (DOE 2004b).

D.1.3.1 Accident Analysis Background

All of the analyzed accidents involve either a release of respirable radioactive material (that is plutonium, uranium, fission products, and activation products as particles and gases) or exposure to direct gamma and neutron radiation. To a lesser extent, accidents may involve the release of fission products from a nuclear criticality. With the exception of uranium and sodium, the quantities of hazardous materials to be handled are small relative to those of many industrial facilities. Consequently, no major chemical accidents are identified, but uranium and sodium are evaluated in terms of hazardous material releases.

For each accident category, a conservative preliminary assessment of consequence is made and, where consequences are significant, one or more bounding accident scenarios are postulated. The building confinement and fire-suppression systems would be adequate to reduce the risks of most spills and minor fires. The systems would be designed to prevent, to the extent practicable, larger fires and explosions. Great efforts have always been made to prevent inadvertent nuclear criticalities, which have the potential to kill workers in the immediate vicinity. In all cases, implementation of a Criticality Safety Program and standard practices are expected to keep the frequency of accidental nuclear criticalities less than or equal to extremely unlikely.

D.1.3.2 Considerations for Accident Scenarios and Frequencies

A range of design-basis and beyond-design-basis accident scenarios has been identified for the VTR and support facilities. For each technology, the process-related accidents possible during operation of the facility have been evaluated to ensure that either their consequences are small or their frequency of occurrence is extremely low. Design features and operating practices would limit the extent of any accidents and mitigate the consequences for the workers, public, and environment. The general categories of process-related accidents considered include the following:

- Drops or spills of materials within and outside the gloveboxes
- Fires involving process equipment or materials, as well as room or building fires
- Explosions initiated by the process equipment or materials or by conditions or events external to the process
- Nuclear criticalities.

The proposed VTR facilities are being designed to meet or exceed the requirements of *Facility Safety,* DOE Order 420.1B or 420.1C (as applicable) (DOE 2005, 2012a). Design-basis and beyond-design-basis natural-phenomenon-initiated accidents are considered for VTR facilities. In accordance with DOE orders and standards, the design of DOE nuclear facilities is based upon the potential consequences of failure due to natural phenomena initiated events. Specifically for the VTR, the critical plant systems for ensuring reactor safety are designed to withstand the highest classifications of natural phenomenon demands. Seismic events are anticipated to impose the greatest demand on the VTR facilities. Because of potential consequences associated with facilities containing plutonium, the facilities are of robust construction.

Because buildings that contain plutonium are of robust construction, few events outside the buildings would have sufficient energy to threaten the building's confinement. The principal concern for events initiated outside the building would be a collision between a large commercial or military aircraft and the facility.

Several facilities evaluated in this VTR EIS have had documented safety analyses (DSAs) prepared. A central focus of the DSA process is to demonstrate that sufficient safety controls are in place, as opposed to quantifying an absolute value of risk. In general, DSAs do not attempt to establish best estimates of the probabilities or consequences of potential accidents. Consistent with their purpose, source terms and other assumptions used for bounding DSA frequency and consequence estimates are conservative. In other words, the DSA process accounts for the inherent uncertainties associated with quantifying risk by requiring that conservative assumptions be made to ensure that the final safety control set is comprehensive and adequate. In contrast, the goal of the accident analysis in this VTR EIS is to present consistent estimates of accident risks between facilities so that fair comparisons can be made among alternatives. If the accident risks between facilities or alternatives are based on differing levels of conservatism, balanced comparisons are not possible.

D.1.3.3 Identification of Material at Risk

For each accident scenario, the radioactive MAR is identified. MAR has a wide range of chemical and isotopic forms. The isotopic composition of MAR will vary, depending on the feed source. Low-enriched uranium (LEU) is also present. For the accidents considered in this VTR EIS, the contribution to dose from LEU releases is a second-order effect when released in conjunction with plutonium. The susceptibility of material to be released under accident conditions generally depends on the form of that material, the degree and robustness of confinement, and the energetics of the potential accident scenario (DOE 1999). For example, plutonium stored in strong, tight storage containers is not generally vulnerable to simple drops or spills, but may be vulnerable in a total structural collapse, as postulated in an earthquake scenario.

Plutonium-239 dose equivalents: The VTR project is considering a wide range of plutonium for use in the production of VTR fuel. Potential fuel sources include excess plutonium metal and oxide stored in K Area of SRS. Plutonium metal might also be available from the excess pit plutonium, excess unirradiated reactor fuel, plutonium from the processing of foreign reactor fuel, or other sources. From an accident perspective, the key differences are the radiological dose impacts from release of a given amount of one material versus another.

Many safety analyses have adopted the strategy of using a convenient surrogate, plutonium-239 dose equivalents, in place of the actual quantities and isotopic composition of the materials. With this approach, the masses or activities of certain quantities of material, such as weapons-grade plutonium, can be expressed in terms of the amount of plutonium-239 that would result in the same radiological dose upon inhalation. Quantities of other materials, such as uranium, can also be expressed in terms of plutonium-239 dose equivalents.

The relative inhalation hazard of source material for VTR fuel production is highly dependent on the amount of each of the plutonium isotopes and on the amount of americium-241 in the mixture. Adding uranium to the fuel mixture is a second-order effect when assessing the inhalation hazard and, therefore, is not considered. Plutonium-241 in the fuel mixture is less of an inhalation hazard than americium-241, but over time, plutonium-241 decays into americium-241 thereby creating an increased inhalation hazard.

To account for the relative inhalation hazard of fuel mixtures, **Table D–1** summarizes the isotopic composition and plutonium-239 dose equivalency of several possible mixtures of plutonium source materials for production of VTR fuel. Dose calculations for accidents evaluated in this VTR EIS use SRS K Area Material Storage Area/K Area Interim Surveillance (KIS)-type plutonium (87.8 percent plutonium-239 and 6.25 percent americium-241) as the reference fuel material. As shown in Table D–1 for the SRS KIS fuel mix, inhalation of 1 gram of the KIS fuel mixture would give the same dose as inhalation of 9.83 grams of plutonium-239.

Table D–1. Potential VTR Plutonium Feed Materials and Plutonium-239 Dose Equivalency

Isotope	*Mid-Range Reactor-Grade Mix* [a]	*Bounding Reactor-Grade Mix* [a]	*Mid-Range Weapons-Grade Mix* [a]	*Bounding Weapons-Grade Mix* [a]	*SPD SEIS SRS KIS Mix* [b]	*Mid-Range Non-Weapons-Grade* [a]	*Bounding Non-Weapons-Grade* [a]
	Weight Fraction						
Pu-238	$2.50{\times}10^{-3}$	$2.10{\times}10^{-2}$	$3.00{\times}10^{-4}$	$5.00{\times}10^{-4}$	$4.00{\times}10^{-4}$	$1.90{\times}10^{-3}$	$9.00{\times}10^{-3}$
Pu-239	$6.94{\times}10^{-1}$	$6.22{\times}10^{-1}$	$9.38{\times}10^{-1}$	$9.29{\times}10^{-1}$	$8.78{\times}10^{-1}$	$8.36{\times}10^{-1}$	$8.30{\times}10^{-1}$
Pu-240	$2.70{\times}10^{-1}$	$2.51{\times}10^{-1}$	$6.01{\times}10^{-2}$	$6.00{\times}10^{-2}$	$1.15{\times}10^{-1}$	$1.48{\times}10^{-1}$	$1.47{\times}10^{-1}$
Pu-241	$8.60{\times}10^{-3}$	$4.00{\times}10^{-2}$	$1.40{\times}10^{-3}$	$1.00{\times}10^{-2}$	$3.70{\times}10^{-3}$	$7.90{\times}10^{-3}$	$7.80{\times}10^{-3}$
Pu-242	$2.54{\times}10^{-2}$	$6.60{\times}10^{-2}$	$4.00{\times}10^{-4}$	$1.00{\times}10^{-3}$	$2.60{\times}10^{-3}$	$6.10{\times}10^{-3}$	$6.10{\times}10^{-3}$
Am-241	$4.68{\times}10^{-2}$	$8.40{\times}10^{-2}$	$4.50{\times}10^{-3}$	$7.00{\times}10^{-3}$	$6.25{\times}10^{-2}$	$2.30{\times}10^{-2}$	$6.50{\times}10^{-2}$
Sum:	1.05	1.08	1.00	1.01	1.06	1.02	1.07
Pu-239 Dose Equivalent (g Pu-239/g mix) [c]	**8.83**	**18.6**	**1.90**	**2.45**	**9.83**	**5.23**	**12.5**
Ratio: Pu-239 Dose Equivalent Mixture/SRS KIS Mix	**0.90**	**1.89**	**0.19**	**0.25**	**1.00**	**0.53**	**1.27**

Am = americium; g = grams; INL = Idaho National Laboratory; KIS = K Area Interim Surveillance; Pu = plutonium; SPD SEIS = Surplus Plutonium Disposition Supplemental EIS (DOE 2015); SRS = Savannah River Site; VTR = Versatile Test Reactor.

[a] SRNL-TR-2020-00171 (SRNL 2020), sum is greater than 1 to bound, from a dose perspective, the mixtures of plutonium.

[b] DOE/NNSA 2012, NEPA Source Document – sum is greater than 1 to bound the K Area mixture of plutonium.

[c] Plutonium-239 dose equivalency based on Federal Guidance Report 13 (EPA 2002).

<u>Hazardous chemicals</u>: The occupational risks of using hazardous chemicals at the VTR and associated facilities are generally limited to material handling and are managed under the required industrial hygiene program. With the exception of uranium and sodium, the quantities of hazardous materials to be used for the VTR project are small relative to those of many industrial facilities. Consequently, no major chemical accidents are identified, but uranium and sodium are evaluated in terms of hazardous material releases.

D.1.3.4 Identification of Material Potentially Released to the Environment

Material must be aerosolized in order to be released to the environment. The amount and particle size of material aerosolized in an accident generally depends on the form of that material and the energetics of the potential accident scenario. Once the material is aerosolized, it must bypass the confinement and filtration systems in order to result in a release to the environment. The types and amounts of radioactive or hazardous material released to the environment following an accident are referred to as the source term.

The standard DOE formula is used to estimate the source term for each accident at each of the proposed VTR facilities:

$$\text{Source Term} = \text{MAR} \times \text{DR} \times \text{ARF} \times \text{RF} \times \text{LPF}$$

where:

MAR = material at risk (curies or grams)
DR = damage ratio
ARF = airborne release fraction
RF = respirable fraction[3]
LPF = leak path factor

MAR is the amount of radionuclides (measured in curies or grams of each radionuclide) available for release when acted upon by a given physical stress or accident. MAR is specific to a given process and potential accident sequence in the facility of interest. It is not necessarily the total quantity of material present; rather, it is that amount of material in the scenario of interest postulated to be available for release.

The damage ratio (DR) is the fraction of MAR exposed to the effects of the energy, force, or stress generated by the postulated event. For the accident scenarios discussed in this analysis, the value of the DR varies depending on the details of the accident scenario but can range up to 1.0.

The airborne release fraction (ARF) is the fraction of material that becomes airborne due to the accident. The respirable fraction (RF) is the fraction of the material with a particulate aerodynamic diameter less than or equal to 10 microns that could be retained in the respiratory system following inhalation. The value of each of these factors depends on the details of the specific accident scenario postulated. ARFs and RFs are estimated according to reference material in *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE 1994).

The leak path factor (LPF) accounts for the action of removal mechanisms (e.g., confinement systems, filtration, and deposition) to reduce the amount of airborne radioactivity ultimately released to occupied spaces in the facility or the environment.

Accident scenarios are identified that result in an airborne release of material. No accident scenarios are identified that would result in a substantial release of plutonium or other radionuclides via liquid pathways. Effects of fission products released during a nuclear criticality accident have been extensively studied. The principal concern is ingestion of iodine-131 via milk that becomes contaminated due to milk cows ingesting contaminated feed. This pathway can be controlled and, in terms of the effects of an accidental criticality, doses from this pathway would be small.

Since it is still early in the VTR design phase, detailed emergency response planning has not occurred. For purposes of the EIS, the radiological impacts are conservatively estimated with the assumption that no emergency planning mitigation measures are taken into account. Thus, the impact estimates do not model sheltering in place, evacuation, and similar measures typically implemented around power reactors. Similarly, for purposes of the EIS, no interdiction or mitigation is assumed to prevent or mitigate long-term exposures to contaminated areas, and foods. But in the event of a large release, such measures would likely occur. Total postulated radiological impact reported includes both the near-term and long-term impacts without mitigation. This conservative approach may result in overestimating health effects within an exposed population.

[3] Respirable fractions are not applied in the assessment of doses based on non-inhalation pathways, such as criticality.

D.1.4 Evaluation of Accident Consequences

D.1.4.1 Potential Receptors

For each potential accident, consequences (in doses) and frequencies are provided for three receptors: (1) a noninvolved worker, (2) the MEI, and (3) the offsite population. The doses are calculated for mean meteorological conditions. Meteorology specific to INL/MFC, ORNL, and SRS is used in the evaluation. Site-specific meteorological data is in the form of hourly readings for wind speed, wind direction, stability class, and rain rate. Dispersion of the materials is highly dependent on the release location, specific weather conditions at the time, and the effects of buildings and other obstructions to enhance or reduce the concentrations.

In practice, there is no way to predict realistically what the actual exposures to the receptors might be. Hence, the DOE safety process uses a prescribed method to calculate impacts and establish safety controls that limit potential radiological impacts on the receptors. Consequences for receptors as a result of plume passage are determined without regard for emergency response measures and, thus, are more conservative than would be expected if evacuation, sheltering, or other measures to reduce or prevent impacts were explicitly modeled. For purposes of this VTR EIS, the hypothetical receptors are assumed to be unaware of the accident and to remain in the plume for the entire passage with no emergency actions taken for protection. Assuming exposure to the entire release maximizes the predicted impacts. If the release of materials from an accident were through a stack, the release would likely be HEPA filtered, thereby reducing the radiological impacts on the receptors. If the release were to occur through a tall stack, the highest exposures could occur several times the stack height downwind. Most of the accidents evaluated for the VTR and support facility operations at INL/MFC and ORNL and fuel production operations at SRS assume failed containment and ground-level releases when evaluating the impacts on the hypothetical receptors. Deviations from the failed containment and ground-level release assumptions are identified in the specific accident evaluations. Given all of these factors, the reported impacts on the receptors are not necessarily realistic, but they are conservative, and application of common assumptions allows a fair comparison among the alternatives.

The first receptor, a noninvolved worker, is a hypothetical individual working on site, but not involved in the proposed activity. The noninvolved worker is assumed to be 330 feet downwind from the accidental releases at the INL/MFC, ORNL, and SRS sites. The evaluation of the potential consequences to a noninvolved worker give some measure of the potential consequences to an onsite worker who might not hear the emergency alarms and take proper protective responses as training would dictate. Realistically, evaluating the potential impacts on a noninvolved worker is highly uncertain. At most DOE sites, including the sites for the proposed VTR and supporting facilities, all workers within the operations area would likely be warned of a potentially significant radiological incident and take proper precautions and response, like taking shelter or evacuating.

The second receptor, an MEI, is a hypothetical individual assumed to be at a location along the site boundary or at a point within the site boundary where a member of the public has unrestricted access. There, this person would be exposed to the entire release and receive the largest dose. Exposures received by this individual are intended to represent the highest potential dose to a member of the public. For the VTR at INL/MFC, the assumed MEI location is on U.S. Highway 20 that runs through the site. The distance from the proposed VTR site to the road is about 3.1 miles. For the VTR at ORNL, the specific location of the VTR and its support facilities is uncertain but the proposed MEI location is about 0.5 miles east of the proposed VTR site on a publicly accessible location on Melton Hill Reservoir. For fuel production operations at the SRS K Area Complex, the MEI is 5.5 miles from the facility.

The third receptor, the offsite population, comprises all members of the public within 50 miles of the accident location. Population projections to 2050 are used to evaluate the population dose for each accident scenario and site. The general public living within a 50-mile radius of the facility and residing directly downwind of the accident receive the maximum exposure via inhalation, ingestion, air immersion, and ground-surface pathways.

For workers directly involved in the processes under consideration, consequences are addressed qualitatively rather than quantitatively. The uncertainties involved in quantifying accident consequences for the involved worker become overwhelming for most radiological accidents due to the high sensitivity of dose values to assumptions about the details of the release and the location and behavior of the affected worker.

D.1.4.2 Modeling of Dispersion of Releases to the Environment

The MACCS2 computer program (WinMACCS, Version 3.11.2) is used to calculate radiation doses and health risks to the noninvolved worker, the maximally exposed offsite individual, and the population within 50 miles of the release point (SNL 2007). SecPop (Sector Population), a U.S. Nuclear Regulatory Commission (NRC) computer program, provides estimates of population, land use, and economic values related to a specific site. It creates a site file that is needed by MACCS2 to perform a site-specific offsite consequence analysis of the health, economic, and environmental impacts of a hypothetical, atmospheric release of radioactive material from a nuclear facility (NRC 2019b). The population for the year 2050 within 50 miles of the approximate midpoint of proposed VTR operations at either the INL/MFC site or ORNL site is estimated using the latest census and past population growth information. A detailed description of the MACCS2 model is available in two NRC documents: *MELCOR Accident Consequence Code System* (NRC 1990) and *Code Manual for MACCS2* (NRC 1998). Originally developed to model the radiological consequences of nuclear reactor accidents, this computer code has been used for the analysis of accidents in many EISs and other safety documentation. A detailed description of the SecPop model is available in NRC's *Sector Population, Land Fraction, and Economic Estimation Program* (NRC 2019b). These computer programs are considered applicable to the analysis of accidents associated with the VTR.

MACCS2 models the consequences of an accident that releases a plume of radioactive materials into the atmosphere. Specifically, it models the degree of dispersion versus distance as a function of historical wind direction, speed, and atmospheric conditions. Were such an accidental release to occur, the radioactive gases and aerosols in the plume would be transported by the prevailing wind, dispersed in the atmosphere, and would expose the population to radiation. MACCS2 generates the downwind doses at specified distances, as well as the population doses out to 50 miles.

To calculate population doses, the region around the facility is divided by a polar-coordinate grid, centered on the facility itself. The user specifies the number of radial divisions and their endpoint distances. The angular divisions used to define the spatial grid correspond to the 16 directions of the compass. Dose distributions are calculated in a probabilistic manner. Releases during each of the 8,760 hours of a year are simulated, resulting in a distribution of dose reflecting variations in weather conditions at the time of the postulated accidental release. The code outputs the conditional probability of exceeding an individual or population dose as a function of distance. The mean consequences are analyzed in this VTR EIS.

The standard MACCS2 dose library is used. This library is based on *Cancer Risk Coefficients for Environmental Exposure to Radionuclides:* Federal Guidance Report 13 (EPA 1999) inhalation dose conversion factors. For exposure to radioactive heavy metal, such as in VTR fuel, the dominant pathway for exposure is inhalation of micron-sized, respirable particles. Absorption through the skin is not a significant pathway for plutonium dose. Overall, the values reported in this EIS are both conservative and internally consistent. The uncertainties in the estimated source terms far outweigh the differences in the modeling and dose conversion factor models used in this EIS.

A key factor in predicting concentration of radioactive particles downwind of a release is the "dry deposition velocity," which is a measure of how fast particles settle to the ground due to gravity. The phenomenon has been well studied and a key factor in the effect of gravity is the size of the particles released. If fine particles (micron or pollen-sized) are released, the settling or deposition velocity is low, and the particles can remain airborne for an extended time. For larger particles, in the range of tens of microns, many of the particles would reach the ground within short distances from the release point. For most types of accidental releases that might occur with the VTR and support facilities, the size of particles that might be released is unknown and speculative. For purposes of this release, a dry deposition of 0.1 centimeters per second is adopted. Using a low value like this likely over predicts the 50-mile population dose, but it is realistic for the MEI dose. This dry deposition velocity is consistent with DOE guidance and recommendations for unmitigated and unfiltered particulate releases for safety analyses using MACCS2 (DOE 2004a). An even more conservative deposition velocity used in some previous DOE EISs of 0 centimeters per second (no deposition) results in 50-mile population doses about 10 percent higher than the 0.1 centimeters per second value used in this EIS.

MACCS2 has the capability of modeling the potential near-term impacts from the initial plume passage and the long-term impacts of the radionuclides remaining after the plume passage. In MACCS2 terms, the near-term phase is called the "early" phase. Because power reactor accidents have the potential for very high radiological impacts during this initial plume phase, and major releases can be delayed for hours or days after the initial event, MACCS2 allows for complicated modeling of the mitigating effects of evacuation of persons within the emergency planning zones around power reactors. For this EIS, the MACCS2 modeling assumptions used for evaluation of initial plume passage or early doses are the default assumptions provided in the MACCS2 sample problem "NRC Point Estimates Linear No-Threshold (LNT)" which evaluates potential power reactor accidents. The MACCS2 LNT input file implements the standard linear, no-threshold, hypothesis model. In the linear, no-threshold model, the relationship between dose and health effect is linear, even for infinitesimal doses. Specification of wet and dry deposition parameters in the input file also serves to achieve conservative calculation results. These assumptions should be conservative and over-predict the initial consequences. Emergency actions, such as evacuation, which could lower the potential public consequences are not modeled.

MACCS2 also allows for the detailed modeling of potential long-term (over decades) radiological impacts from the radiological materials that remain in the surrounding area after initial plume passage. In MACCS2, this long term is called the "chronic" phase and includes the combined effects of direct exposure (inhalation) to residual particulates, ingestion of contaminated foods, radiation exposure from residual material on the ground (ground shine), inhalation of disturbed, residual ground-level particulates (resuspension), and ingestion of contaminated water, as applicable. This modeling approach has evolved primarily to evaluate the potential long-term impacts of nuclear power plant accidents, such as Chernobyl and Fukushima. Due to the nature of the problem of modeling long-term consequences, there are great uncertainties and many site-specific differences that affect many of the calculations. Many of the modeling assumptions require detailed, site-specific understanding of local food growth and consumption patterns to estimate food and water dose; local soil conditions to predict resuspension; and an array of variables to predict chronic direct and ground-shine exposures. For these reasons, the estimates of long-term doses or chronic doses should be viewed with caution and the recognition that they are highly uncertain. For this EIS, the MACCS2 modeling assumptions used for evaluation of chronic doses are the default assumptions provided in the MACCS2 sample problem "NRC Point Estimates LNT," which evaluates potential power reactor accidents. These assumptions should be conservative and over-predict the long-term consequences.

As implemented in this EIS for accidents at VTR facilities, the MACCS2 model evaluates doses due to inhalation of aerosols containing respirable radionuclides and to direct exposure from radionuclides in the passing plume. This model represents the major portion of the dose that a noninvolved worker or member of the public would receive from a VTR or support facility accident. The long-term effects from exposure to radionuclides deposited on the ground and surface waters, from resuspension and inhalation of radionuclides, and from ingestion of contaminated crops also are modeled. These long-term pathways have been studied and found not to contribute as significantly to dose as inhalation, and they would be controllable through interdiction. For purposes of this EIS, both the near-term (early) and long-term (chronic) impacts are reported.

Long-term effects of fission products released during a nuclear criticality, reactor accidents, and accidents involving used or spent fuel after removal from a reactor have been extensively studied. For reactor accidents, ingestion of contaminated food can be the major pathway for long-term exposure. For accidents involving reactor fuel, ingestion of isotopes of cesium, strontium, and iodine can result in exposures much larger than the initial exposures during plume passage. As such, mitigation measures are often planned for power reactors to reduce the potential public impacts. The modeling results in this EIS report the long-term doses to the population within 50 miles.

Radiological consequences may vary somewhat because of variations in the duration of release. For longer releases, there is a greater chance of plume meander caused by variations in wind direction over the duration of release. MACCS2 models plume meander by increasing the lateral dispersion coefficient of the plume for longer release durations, thus lowering the centerline dose reported by MACCS2. For perspective, centerline doses from a homogenous 1-hour release would be 30 percent lower than those from a 10-minute release because of plume meander. Centerline doses from a 2-hour release would be 46 percent lower. The other effect of longer release durations is involvement of a greater variety of meteorological conditions in a given release, which reduces the variance of the resulting dose distributions. This would tend to lower high-percentile doses, raise low-percentile doses, and have no effect on the mean dose.

A duration of 10 minutes is assumed for all VTR facility accident releases. This is consistent with the accident phenomenology expected for all scenarios, with the possible exception of fire. Depending on the circumstances, the time between fire ignition and extinction may be considerably longer, particularly for the larger beyond-design-basis fires. However, even in a fire of long duration, it is possible to release substantial fractions of the total radiological source term in short periods as the fire consumes areas of high MAR concentrations. The assumption of a 10-minute release duration for fire is intended to represent this circumstance.

D.1.4.3 Long-Term Consequences of Releases

The probability coefficients for determining the likelihood of fatal cancer, given a dose, are taken from the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991) and DOE guidance (DOE 2004b). For low doses or low dose rates, probability coefficients of 6.0×10^{-4} fatal cancers per rem and person-rem are applied for workers and the general public, respectively (DOE 2003). For cases where the individual dose would be equal to or greater than 20 rem, the LCF risk is doubled (NCRP 1993). For severe accidents where members of the public could receive more than 20 rem, the total and early cancers from the MACCS2 calculations are reported. Additional information about radiation and its effects on humans is provided in Appendix C.

D.2 Considerations for Accident Selection

D.2.1 Background

High-energy events would be expected to damage some of the confinement barriers provided in the facility design and could remove their ability to provide mitigation of potential releases. Medium-energy events would be expected to reduce the effectiveness of the barriers but would not be expected to defeat them. However, low-energy events would have almost no impact on the ability of the confinement barriers to perform their function. A review of the accident scenarios indicated that only severe accident conditions (e.g., accidents involving confinement failure) could result in a significant release of radioactive material to the environment or an increase in radiation levels. These severe accident conditions are associated with beyond-design-basis events, combinations of events for which the facility is not specifically designed. While these events would be expected to have consequences larger than those associated with design-basis events, their frequency would be expected to be much lower than the design-basis event frequency. Natural phenomena (e.g., earthquake) events and fire accidents creating a direct path for releases to the environment represent the situations with the highest consequences to the public. Some types of events, such as procedure violations and spills of small quantities of material containing radioactive particles and most other types of accidents caused by human error, occur more frequently than the more severe accidents analyzed. However, these accidents do not involve enough radioactive material to result in a significant release to the environment, although the impact on operational personnel may be significant. The airborne particles from a process-related accident would normally pass through at least one bank and possibly two to four banks of HEPA filters before entering the environment. Spent nuclear fuel handling operations are performed inside confinement barriers, such as hot cells or canyon walls. The hot cells are equipped with significant safety features, such as flow and pressure control and in some cases, an inert gas atmosphere. These features are modeled in the analysis when their operability is not compromised by the sequence of events associated with the accident progression. While severe accidents (also referred to as beyond-design-basis events) are expected to have the most significant impact, that is, the highest consequences on the population, these accidents may not have as significant a risk impact on all receptors as higher-frequency, lower-consequence accidents. For this reason, higher-frequency accident scenarios are included in the accident analysis.

D.2.2 Accident Scenario Consistency

In preparing the accident analysis for this VTR EIS, the primary objective is to ensure consistency in the accident analyses so that the results of the analyses for the proposed VTR and supporting facility alternatives and options impacts can be fairly compared. The accidents selected for analysis provide a consistent basis for comparing the alternatives and options in this EIS. The following sections discuss various accident categories.

Aircraft crash. Frequencies of an aircraft crash into each facility evaluated in this VTR EIS under each alternative are developed in accordance with the *Accident Analysis for Aircraft Crash into Hazardous Facilities* (DOE 2006). External events, such as the crash of a large aircraft into a VTR facility structure with an ensuing fuel-fed fire, are conceivable. At most locations away from major airports, however, the likelihood of a large aircraft crash is less than 1 in 10 million per year (1×10^{-7} per year). In most cases, the building is considered to provide sufficient structural strength and shielding such that a release of radioactive material would not be likely.

Inadvertent Nuclear Criticality. The source term for an inadvertent nuclear criticality is based on a fission yield of 1.0×10^{19} fissions, which is used for all facilities analyzed in this VTR EIS. The source term is based on that given in *DOE Handbook: Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE-HDBK-3010-94) (DOE 1994). The estimated frequency of an inadvertent nuclear criticality is often assigned to the extremely unlikely bin to be conservative.

Design-basis earthquake. All of the existing and proposed facilities that are considered in this VTR EIS would be expected to have seismic evaluations demonstrating that they meet the seismic evaluation requirements for a design-basis earthquake.

Beyond-design-basis earthquake. All of the proposed operations would be in either existing or new facilities that are designed to meet or exceed the requirements of DOE Order 420.1B or 420.1C (DOE 2005, 2012a), as applicable. Facilities would also meet applicable requirements of DOE-STD-1020, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities* (DOE 2002, 2012b, 2016) for reducing the risks associated with natural phenomenon hazards. The frequency of beyond-design-basis earthquakes for all facilities may be reported in this VTR EIS as extremely unlikely to beyond extremely unlikely. The proposed facilities would be characterized as Natural Phenomena Hazard (NPH)

Design Category 3 (NDC-3) or Performance Category 3 (PC-3) for fuel production and VTR experiment hall facilities and NDC-5 for the VTR reactor and supporting safety class systems.[4]

The numerical seismic design requirements detailed in DOE-STD-1020 are structured such that there is assurance that specific performance goals would be met. For PC-3, NDC-3, and NDC-5 designated facilities, the performance goal is to ensure occupant safety and hazard confinement for earthquakes with an annual probability of occurrence exceeding approximately $4{\times}10^{-4}$, $1{\times}10^{-4}$, or $1{\times}10^{-5}$, respectively. There is sufficient conservatism in the design of the buildings and the structures, systems, and components that are important to safety that this goal should be met, given that they are designed to withstand earthquakes with an estimated mean annual probability of $4{\times}10^{-4}$, $1{\times}10^{-4}$, or $1{\times}10^{-5}$.

By contrast, nonnuclear structures at these sites and the surrounding community would be constructed to the regional standards of the *Uniform Building Code* or *International Building Code* at the time of construction. Specifically, peak acceleration values are 50 to 82 percent of the design requirements for nuclear facilities in the same area and correspond approximately to DOE PC-1 or PC-2 facilities with 500-year return intervals. During major earthquakes, structures built to these *Uniform Building Code* or *International Building Code* requirements are expected to suffer significantly more damage than reinforced structures designed for nuclear operations.

For the VTR, a site-specific probabilistic seismic hazard analysis and probabilistic risk assessment will be performed to characterize the site for potential seismically-induced ground motions with ground motion magnitude and frequency estimates for return periods up to 100,000 years. The details of potential earthquakes with return periods in this range is uncertain and uncertainty bounds and estimates are accounted for in the definition of the seismic hazard, which governs the system design and analyses. Even though details of seismic events with return periods greater than 100,000 years are uncertain, some evaluations of facility margins to these events can be assessed based upon the ratio of NPH-induced system stresses to their actual design capacity. For purposes of this VTR EIS, it is assumed that, at all the candidate sites, earthquakes with return periods in the 10,000- to 10-million-year range might result in sufficient ground motion to cause major damage to the facilities classified as SDC-3 or PC-3. The level of damage necessary for failure of the SDC-5 structures would be so large that the overall infrastructure damage and consequences of such an event to the nearby area infrastructure would likely be catastrophic negating any VTR consequences. Such catastrophic damage is evidenced by the results of the 2011 Tohoku earthquake in Japan, which destroyed many homes and much infrastructure. However, the nearby reactors, which were designed to be equivalent of SDC-5, performed as expected and designed until impacted by the tsunami. Neither of the potential VTR candidate sites are subject to tsunami hazards. VTR reactor safety systems are designed to the most rigorous construction standards and will be evaluated to ensure that the annual probability of seismically-induced failure is outside the frequency of concern when considering the frequency of the events and the component fragilities of the key safety systems.

Filtration efficiency. This VTR EIS analysis assumes the exhaust from most facilities, including the existing K-Reactor and MFC facilities, as well as the proposed ORNL Post-Irradiation Examination/Spent Fuel

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[4] Each structure, system, and component in a DOE facility is assigned to one of five NPH Design Category categories, or to one of four performance categories, depending on its safety importance. Structures, systems, and components designated as SDC-3 or PC-3 or higher are those for which failure to perform their safety function could pose a potential hazard to public health, safety, and the environment from release of radioactive or hazardous materials. Design considerations for this category are to limit facility damage as a result of design-basis natural phenomena events (for example, an earthquake) so that hazardous materials can be controlled and confined, occupants are protected, and the functioning of the facility is not interrupted (DOE 2002, 2012c, 2016). The return periods or other criteria under which the design and analysis needs to demonstrate confinement of radioactive or hazardous material are progressively higher as the design category goes from a 1 to a 5 and are set based upon estimates of the public and worker consequences as a result of failure caused by the specific NPH.

facility, would be directed through two stages of testable HEPA filters to a stack. A building LPF of 1.0×10^{-5} is used for particulate releases with HEPA filters unless otherwise noted (DOE 1999). Under design-basis accident conditions, the HEPA filters are assumed to remain functional but with degraded efficiency and a building LPF of 5×10^{-3} is assumed. Under more severe conditions, such as major fires, the HEPA filters are assumed for purposes of this EIS to be severely degraded. Under these severely degraded conditions, a building LPF of 1×10^{-1} is assumed. Under beyond-design-basis conditions where structural failure of a building is assumed, a building LPF of 1 is conservatively assumed.

For the hypothetical beyond-design-basis earthquake and fire accident scenarios, a consistent LPF is assumed across the facilities evaluated. In this VTR EIS, the hypothetical beyond-design-basis earthquake accidents are not based on detailed analysis, and are postulated simply to show a bounding level of impacts should the safety design and operational controls fail. For NEPA purposes, the goal is to show the impacts of realistic, physically possible events even if it is believed their probability is extremely low.

For comparison purposes, it is postulated that:

- The hypothetical beyond-design-basis accident is an earthquake that exceeds the design-basis earthquake (for example, SDC-3 for the experiment hall and SDC-5 for the VTR) by a sufficient margin that gloveboxes fail, fire suppression systems fail, power fails, and some building confinement is lost. It is further assumed that a room-wide fire or multiple local fires might occur. The overall probability of the event, considering the conditional probabilities of fires following a beyond-design-basis earthquake, is expected to be in the 1×10^{-6} to 1×10^{-7} per year range or even lower.
- For new facilities and significantly upgraded facilities, a building LPF of 0.1 could be assumed and expected to be conservative. This factor should adequately represent an LPF for cracks in the building or transport through rubble. Nevertheless, this VTR EIS assumes an LPF of 1 for these facilities even though the LPF could be several times lower than this.
- For older, existing facilities that have not been or are not planned to be upgraded, it is not generally known how they might fail in a beyond-design-basis earthquake but an LPF of 1 is considered unrealistic because even a rubble pile in a total building collapse offers some impediment to particulates being released to the environment. Nevertheless, this VTR EIS assumes an LPF of 1 for these facilities even though the LPF could be several times lower than this.
- For all facilities, an LPF of 1 is assumed for gaseous releases.

The real-world performance of multiple stages of HEPA filters has been well demonstrated and experimental testing confirms the performance of HEPA filters for uranium and plutonium particles. The independent Defense Nuclear Facilities Safety Board (DNFSB) thoroughly evaluated the use of HEPA filters by DOE and issued multiple reports on the performance of HEPA filters within the DOE complex. HEPA filters used in support of the VTR activities would conform to the latest version of DOE Standard "Specifications for HEPA Filters Used by DOE Contractors," DOE-STD-3020-2015. Performance testing required by this standard for all HEPA filters installed for safety would ensure that the filters meet or exceed the performance requirements assumed in safety evaluations.

D.2.3 Accident Scenario Differences among Alternatives and Options

The accident scenarios defined for the different alternatives and options have the same initiating events and scenario descriptions. The differences among the alternative are due to different distances to the noninvolved worker and MEI, different population distributions, different meteorological conditions, and different release heights for elevated releases. Other parameters for radionuclide inventory, release factors, release duration, wet and dry deposition, and food ingestion factors are the same in a set of accidents for the alternatives and a set of accidents for the options.

D.2.3.1 INL VTR Alternative

At INL, the VTR would be built adjacent to and east of the currently protected area at MFC. Fuel manufacturing would occur in the Fuel Manufacturing Facility (FMF) and Zero Power Physics Reactor (ZPPR) protected area at the MFC. Post-irradiation examination and spent fuel treatment would occur in the Hot Fuel Examination Facility (HFEF) and the Fuel Conditioning Facility (FCF). Spent fuel would be stored in casks on the spent fuel storage pad (Marschman et al. 2020).

D.2.3.2 ORNL VTR Alternative

At ORNL, the VTR and a new Hot Cell Building would be built on land approximately a mile east of the High Flux Irradiation Reactor (HFIR) complex. Fuel manufacturing would not be performed at ORNL. Post-irradiation examination and spent fuel treatment would occur in the Hot Cell Building. Spent fuel would be stored in casks on the spent fuel storage pad.

D.2.3.3 VTR Fuel Production Option at Savannah River Site

Under the SRS fuel production option, fuel production would be performed in the K-Reactor Building (105-K) in the K Area Complex. All equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and fuel assembly fabrication would be located on two below-ground levels within the building. An induction-casting furnace would be used in the initial steps in the fuel fabrication process, alloying the elemental metallic components and producing the fuel slugs. This furnace would be contained within a glovebox with an inert gas atmosphere. Under the SRS feedstock preparation option, this capability would be located adjacent to the location for the fuel fabrication capability, in the K-Reactor Building (105-K) in the K Area Complex.

D.2.3.4 VTR Fuel Production Option at Idaho National Laboratory

The INL Fuel Production Option includes the use of the FMF and the ZPPR to house the equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and fuel assembly fabrication. All of the equipment for feedstock preparation would be newly constructed. VTR fuel production is projected to require sample analysis for hundreds and potentially thousands of samples in the first few years of operation. INL proposes to use existing space fitted with new equipment in the FCF as an analytical chemistry laboratory to support VTR fuel production. Like the SRS option, an induction-casting furnace at INL would be used in the initial steps in the fuel fabrication process, alloying the elemental metallic components and producing the fuel slugs. This furnace would be contained within a glovebox with an inert gas atmosphere.

D.2.3.5 No Action Alternative

Under the No Action Alternative for the VTR and the No Action Option for fuel production, no new facilities would be built or modified at INL, ORNL, or SRS. Existing test reactors and support facilities would continue operating under either the no action or action alternatives and options, so there would be no incremental change to the associated impacts. Impacts of continued operation are included in those representing the existing affected environment.

D.3 Facility Accident Scenarios

D.3.1 Reactor Fuel Production Accidents at Idaho National Laboratory and Savannah River Site

As discussed in Section D.1.3.3, the VTR project is considering a wide range of plutonium for use in the production of VTR fuel. Radiological impact calculations for VTR feedstock preparation and fuel fabrication activities at SRS or INL assume that the fuel material is from the KIS mixture of plutonium oxide. This is a hypothetical mixture of plutonium developed for safety basis analysis at the K-Reactor Area and is expected to adequately represent the plutonium materials stored in K Area. From an accident perspective, the key differences among accident scenarios are the radiological dose impacts from release of a given amount of one plutonium mixture versus another. The differences in radiological impacts of one mixture of plutonium versus another are partially due to the build-up of americium-241 from the decay of plutonium-241. To show the effect of one plutonium mixture versus another, the radiological dose impacts of each potential plutonium type have been evaluated and compared to the radiological dose impacts of the KIS mixture of plutonium oxide as shown in Table D–1. Table D–1 provides a summary of potential sources of plutonium for VTR fuel production, the isotopic composition of the fuel, the plutonium-239 dose equivalency of the fuel, and the ratio of plutonium-239 dose equivalency of the fuel to the SRS KIS mixture of plutonium oxide.

The reference metallic fuel (consisting of an alloy of uranium, plutonium, and zirconium) to be used in the VTR is unique and would be fabricated at either the SRS or the INL MFC. Materials available for use in the production of the metallic fuel (feedstock) exist in several forms. Plutonium feedstock may be in the form of metals or oxides while uranium feedstock may be in metal form.

Depending on the feed material, additional processing of the plutonium may be necessary. Additional processing could include:

- Conversion of plutonium oxide to metal
- Removal of gallium from plutonium metal to improve desirability for reactor fuel
- Removal of americium-241, a significant source of radiological extremity dose to glovebox workers.

The fuel form for the fuel pin is a cast metallic cylindrical slug. Depending on the form of the feedstock, the steps needed to fabricate the fuel slug would be:

- Feed material polishing, removal of impurities;
- Conversion of feedstock from non-metallic forms to metals;
- Fuel alloying and homogenization;
- Fuel slug casting and demolding;
- Assembly of the fuel slugs into fuel pins; and
- Assembly of the fuel pins into fuel assemblies.

The fuel production facility would need to provide space for material storage. Materials to be stored include:

- Plutonium feedstock;
- Uranium feedstock;
- Zirconium feedstock;
- Fuel slugs;
- Fuel pins;
- Fuel cladding;
- Sodium;
- Fuel assemblies; and
- Scrap and waste.

Under the SRS fuel production option, fuel fabrication would be performed in the K-Reactor Building in the K Area Complex. All equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and fuel assembly fabrication would be located on two below-ground levels within the building. The fuel production facility would be located on the minus-20 and minus-40-foot levels (20 and 40 feet below grade) of the K-Reactor Building. Approximately 11,500 square feet and

12,000 square feet of space would be made available at the minus-40 and minus-20 levels, respectively. The facility could support feed material purification, ingot manufacturing, and/or the fabrication of fuel from ingots. New equipment would be provided for fuel slug casting, slug trimming and inspection, fuel rod loading and inspection, fuel bundle assembly and packaging, and waste handling. Other infrastructure to be supplied would include material storage areas, special nuclear material measurement equipment, analytical support, and other infrastructure services, such as glovebox and room ventilation and electrical distribution. Because SRS is not a proposed site for the VTR, completed assemblies (or fuel pins if assemblies are fabricated at the reactor site) would be loaded into a shielded transfer cask and then loaded into a shipping container for shipment.

The INL MFC fuel production option includes the use of the existing FMF and ZPPR to house the equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and fuel assembly fabrication. Under this option, the ingots of each fuel component (uranium, plutonium, and zirconium) would be delivered to INL. The FMF has capabilities to produce and purify transuranic and enriched-uranium feedstock, to develop transuranic metallic and ceramic fuels and to store these fuels. Space for lag storage of casting scrap, and assembled fuel pins pending transfer to ZPPR would be made available in the FMF vault. Additional facilities at MFC would be required to support fuel preparation (conversion of plutonium oxide to plutonium metal, and plutonium processing ("polishing") to remove undesirable impurities).

Activities at either SRS or MFC could result in accidents involving the fuel production. Fuel production activities would be conducted in gloveboxes. Because of the design of the gloveboxes, fuel production accidents would not be expected to result in radioactive or hazardous material releases to the room. Fuel production accidents would result primarily in an elevated release of materials. The material would be released to the glovebox and then exhausted through HEPA filters and out of a stack. Accident scenarios potentially relevant to the proposed fuel production activities are discussed in the SRS and INL Data Reports (INL 2020a; SRNS 2020).

In addition to specific fuel production activities, other potential activities at the SRS and INL fuel production facilities include conversion of plutonium oxide to plutonium metal, and processing ("polishing") the plutonium oxide or metal to remove undesirable impurities, specifically gallium and americium-241. The specific polishing process that would be at SRS and INL has not been finalized at this time but an aqueous process similar to that used for the mixed oxide (MOX) plant and a pyrochemical process similar to the electrorefining process at INL are both potential processes.

The MOX plant at SRS had a front-end polishing process to remove impurities from the plutonium oxide feed material (DOE 2015; NRC 2005; ORNL 1998). The "aqueous polishing" process used in the MOX plant is described and evaluated in detail in the *Environmental Impact Statement on the Construction and Operation of a Proposed Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina* (MFFF EIS) (NRC 2005). The MOX facility was designed (at least initially) to process up to 3.5 metric tons of plutonium annually so its throughput was likely a factor of ~7 times the VTR needs. The MOX aqueous polishing process thus had about 7 times the annual throughput needed for the VTR fuel production facility but the technology and accident scenarios could be similar.

Principal accidents identified in the MFFF EIS (NRC 2005) and the DOE *Surplus Plutonium Disposition Supplemental Environmental Impact Statement* (SPD Supplemental EIS) (DOE 2015) include spills or fires involving plutonium solutions. Two relevant accident scenarios are identified in the DOE SPD Supplemental EIS (DOE 2015) for the aqueous polishing portion of the MFFF. The scenarios involved a leak of liquid organic solvent containing the maximum plutonium concentration (a MAR of 40 grams of plutonium) and a liquid spill of concentrated aqueous plutonium solution (a MAR of 5,000 grams of plutonium). In addition, ORNL 1998 identified additional bounding accidents, including a thermal excursion in an ion exchange column. The postulated thermal excursion in an ion exchange column resulted in the highest potential release, as evaluated in this EIS.

The 2015 SPD Supplemental EIS (DOE 2015) evaluated a range of potential accidents in K Area associated with handling 3013 cans of plutonium oxide. The highest consequence operational or natural phenomena-initiated accidents are all in the extremely unlikely or lower frequency range and involved over-pressurization of a DOE standard 3013 can of plutonium oxide. These containers are extremely robust and consist of two welded, nested stainless-steel containers. Because of their robust construction, they can pressurize to the point of rupture if subjected to a fire of sufficient magnitude and length. This could cause a high-pressure release of plutonium oxide. For the oxide-to-metal conversion processes, all of the operations would be criticality-safety limited, so MAR also would be limited. No accidents involving an oxide-to-metal conversion process (spills, fires, explosions, etc.) would result in a higher amount of material becoming airborne than the over-pressurization of the 3013 can of oxide.

To evaluate the possible effects of fuel production accidents, a criticality and several fires were evaluated. In addition, potential accidents associated with converting plutonium oxide to metal and "polishing" operations are discussed. Polishing operations, whether aqueous or pyrochemical, would result in a high americium-241 content waste stream. This waste stream could ultimately be converted to a solid form for disposal as high-activity TRU waste. The evaluation discusses the release factors used to determine the source term for the accident. An LPF of 1 is used to model the effects when confinement fails. Under severe accident conditions, such as fires, the HEPA filters are assumed, for purposes of this EIS, to be degraded. Under these degraded conditions, a building LPF of 0.1 is assumed to model the accident effects when confinement is intact. The evaluation also considers hazardous material releases.

D.3.1.1 Criticality while Melting Plutonium Metal and Adding Uranium and Zirconium (Fuel Received as Plutonium Metal)

An inadvertent nuclear criticality is assumed to occur in the fuel fabrication glovebox line while melting plutonium metal and adding LEU and zirconium. The criticality results from a seismic event that also causes failure of confinement. The source term for this criticality is based on a fission yield of 1.0×10^{19} fissions. This source term, which is used for all facilities, is based on that given in DOE-HDBK-3010-94 (DOE 1994). Criticality safety controls should prevent this accident from occurring and the materials involved should remain at less than a critical mass. Sufficient acceleration to cause this much damage would require an earthquake with accelerations higher than the design-basis requirements for the structure and failures of the building and equipment would be expected. Thus, this accident is identified as extremely unlikely. The scenario represents a metal criticality. The metal is postulated to soften during the process, resulting in a 100 percent release of gaseous fission products generated in the criticality. However, the scenario does not assume release of aerosolized, respirable metal fragments. Engineered and administrative controls would be available to ensure that the double-contingency principles are in place for all portions of the process. For purposes of this VTR EIS, the DR is 1, and the bounding ARF and RF value of 1 is assumed for noble gases. No radionuclide deposition during transport through the building or the accident debris is assumed, so an LPF of 1 is modeled. The plume rise option is not activated for this scenario. Because the building structure remains intact, a stack release is modeled. **Table D–2** summarizes the MAR, release fractions, and source terms for VTR fuel fabrication facilities.

D.3.1.2 Fire Impingement on Fuel Material (Intact Confinement)

In this scenario, a fire would affect the solid fuel material in a glovebox in the fuel fabrication line. The plume rise option is not activated for this scenario. Because the building structure remains intact, a stack release is modeled. The frequency of the fire is extremely unlikely. MAR is assumed to be 5,000 grams of KIS plutonium mixture. All material in the fuel fabrication line would be at risk, and the DR is 1. The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively, are based on the airborne release of plutonium particulates formed by oxidation at elevated temperatures without self-sustained oxidation (DOE 1994). Reduction in the source term due to the fuel fabrication line confinement is modeled (LPF=0.1). Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.3 Fire impingement on Fuel Material with Seismically-Induced Confinement Failure

The fire selected for analysis would affect the solid fuel material in a glovebox in the fuel fabrication line. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. This much damage would require an earthquake with accelerations substantially higher than the design-basis requirements for the structure and major failures of the building and equipment would be expected. The frequency of the earthquake is extremely unlikely to beyond extremely unlikely. MAR is assumed to be 5,000 grams of KIS plutonium mixture. All material in the fuel fabrication line would be at risk, and the DR is 1. The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively, are based on the airborne release of plutonium particulates formed by oxidation at elevated temperatures without self-sustained oxidation (DOE 1994). A building LPF of 1 is assumed, although a more realistic value is likely to be at least a factor of several lower. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure

A seismically-induced spill of molten plutonium-uranium mixture while heating or casting is assumed to occur. The spilled material is then postulated to rapidly oxidize or burn. This much damage would require an earthquake with accelerations substantially higher than the design-basis requirements for the structure and major failures of the building and equipment would be expected. The frequency of the earthquake is extremely unlikely to beyond extremely unlikely. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. For this accident, a MAR of 4,500 grams of KIS plutonium mixture, a DR of 1, an ARF of 5×10^{-4}, and an RF of 0.5 are estimated, which would result in a release of 1.1 grams to the building. A building LPF of 1.0 is assumed, although a more realistic value is likely to be at least a factor of several lower. Using this LPF would result in a release of 1.1 grams of plutonium mixture to the environment. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.5 Plutonium Oxide-to-Metal Conversion – Explosion of 3013 Container of Plutonium Oxide

The bounding mitigated explosion event identified in the 2015 SPD Supplemental EIS (DOE 2015) for K Area plutonium oxide handling activities is a postulated deflagration or detonation in the glovebox that occurs just as a 3013 container is being punctured for sampling purposes. This accident scenario is applicable to both K Area and MFC if oxide would be used as a feedstock material. The plume rise option is not activated for this scenario. Because the building structure remains intact, a stack release is modeled. The EIS indicates that the internal pressure should be within the 3013-container design rupture limit of 700 pounds per square inch (gauge) unless subjected to an external fire. For this pressure, the expected ARF × RF is 0.022. This ARF × RF yields approximately 99 grams of plutonium mixture from a drum containing 4,500 grams (88.4 percent plutonium from 5,090 grams plutonium oxide, 11.6 percent oxygen) of KIS plutonium mixture. Given a scenario where that mixture is released to the building exhaust system, the building HEPA filters would reduce the amount released to the stack. A building LPF of 0.1 is assumed for one stage of HEPA filters. Therefore, the mitigated release to the environment through the stack would be approximately 9.9 grams of plutonium mixture or 97.3 grams of plutonium-239 equivalent. A release of this magnitude would fall in the extremely unlikely category. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.6 Beyond-Design-Basis Fire Involving a Transuranic Waste Drum Fire

A beyond-design-basis fire has been postulated in the K Area Complex and for MFC that would involve an unmitigated TRU waste drum fire on the loading dock that burns with sufficient intensity and duration that all of the material in the drum is consumed. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The expected ARF x RF is 0.0005, which corresponds to approximately 0.2 grams (0.007 ounces) of plutonium from a drum containing 398 grams (14 ounces) of plutonium mixture (88.4 percent plutonium from 450 grams of plutonium oxide, 11.6 percent oxygen). Because this fire is postulated to occur outside the building, an LPF of 1 is assumed. This accident is conservatively estimated to have a frequency of extremely unlikely to beyond extremely unlikely. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.7 Aqueous/Electrorefining Fuel Preparation

Fires, spills, and explosions for the aqueous plutonium polishing are evaluated. The scenario is based on an analysis for MFFF (ORNL 1998). The following are the dominant accident scenarios identified (ORNL 1998).

Fire. It is assumed that the liquid organic solvent leaks as a spray into the glovebox, builds to a flammable concentration, and is contacted by an ignition source. A MAR of 1 liter of solvent containing 40 grams of plutonium is postulated (ORNL 1998). The combined ARF and RF value for this scenario is 1.0×10^{-2} for quiescent burning to self-extinguishment (DOE 1994; ORNL 1998). A release of 0.4 grams of plutonium to the glovebox is postulated. Under these accident conditions, a building LPF of 0.1 is assumed for one stage of HEPA filters. The frequency of this accident is in the unlikely range.

Spill. Leakage of liquids from process equipment must be considered as an anticipated event. However, with multiple confinement barriers, a release from the process room would be extremely unlikely. A bounding scenario involves a liquid spill of concentrated aqueous plutonium solution (100 grams per liter plutonium) (ORNL 1998), with 50 liters containing 5,000 grams of plutonium accumulating before the leak is stopped. The ARF and RF values used for this scenario are 2.0×10^{-4} and 0.5, respectively (DOE 1994; ORNL 1998). A release of 0.5 grams of plutonium to the glovebox is postulated. A building LPF of 5.0×10^{-3} is assumed for one stage of HEPA filters (ORNL 1998; DOE 2015).

Uncontrolled Reaction/Explosion. The highest consequence operational accident identified by ORNL for the aqueous polishing portion of MFFF is a thermal excursion in a nitrate anion exchange column (ORNL 1998). This scenario examines the potential effects of a thermal excursion within an ion exchange column while the aqueous process is operating. The thermal excursion is postulated to result from off-normal operations, degraded resin, or a glovebox fire. It is also assumed that the column venting/pressure relief fails to vent the overpressure causing the column to rupture violently. The overpressure releases plutonium nitrate solution as an aerosol within the affected glovebox that in turn is processed through the ventilation system. If the overpressure also breeches the glovebox, a fraction of the aerosol will be released within the room as well.

The total mass of KIS plutonium mixture that could be contained in an ion exchange column is 1,000 grams on the resin and 246 grams in nitrate solution. These quantities are based on the maximum intended plutonium loading of the resin and the maximum intended plutonium concentration in solution after pH-adjustment, respectively. DOE 1994 lists ARF/RF values of 9×10^{-3} for burning resin and 6×10^{-3} for liquid behaving as a flashing spray upon depressurization. ORNL 1998 assumes 10 percent of the resin is assumed to burn upon release, or a DR of 0.1. Thus, the release from the resin fire is 1,000 grams × 0.1 × 9×10^{-3} or 0.9 grams of plutonium. The release from the liquid ejected from the column because of overpressure from the flashing solution is 246 grams × 1 × 6×10^{-3} = 1.5 grams of plutonium. An electrorefining process is assumedly operating while the aqueous process is operating. In the electrorefining process, the chopped fuel is placed in anode basket(s) and a current passing between the anode(s) and cathode(s) anodically dissolves the actinides into the molten chloride salt at 1,202 degree Fahrenheit. Multiple cathodes at different electric potentials allow deposition of TRU isotopes on different cathodes. The ARF and RF values used for the release from the electrorefiner are 1×10^{-3} and 0.4 for heavy metal salts (DOE 1994:3-25). The total mass of KIS plutonium mixture that could be contained in the electrorefiner is 3,500 grams. A DR of 1 is assumed. The release from the electrorefiner into the glovebox is 1.4 grams of plutonium (3,500 grams $\times 1 \times 1\times10^{-3} \times 0.4$). Summing these three airborne source terms from the resin, the solution, and the electrorefiner gives a total of 3.8 grams of plutonium mixture aerosol in the glove box. A building LPF of 5.0×10^{-3} is assumed for one stage of HEPA filters. The plume rise option is not activated for this scenario. Because the building structure remains intact, a stack release is modeled. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.8 Aircraft Crash into VTR Fuel Production Facility

A crash of a large, heavy commercial or military aircraft directly into the fuel production facility might damage the structure sufficiently to breach confinement and disperse material into the environment. A subsequent fuel-fed fire could provide energy to further damage structures and equipment, aerosolize material, and drive materials into the environment. Source terms are highly speculative but could be similar to those from the beyond-design-basis earthquake if the structure is not sufficiently robust.

The probabilities of aircraft crashes into SRS facilities, including the K-Reactor Building, have been previously evaluated (Radder 2006) and reported in the K Area DSA (SRNS 2016) and the SPD Supplemental EIS (DOE 2015). The K Area DSA dismisses events initiated by a large or military aircraft because those were shown to be beyond extremely unlikely. The K Area DSA includes evaluation of small aircraft/helicopter (security) crashes since those probabilities are in the extremely unlikely range.

At MFC, the feedstock preparation activities would occur in the FCF followed by fuel fabrication within the FMF and the ZPPR. Due to the remote location of MFC, the probability is sufficiently low that impacts from aircraft crashes are not addressed in the facility safety documents. For purposes of this VTR EIS, MAR within these facilities is assumed vulnerable to a direct aircraft impact. An aircraft impact into the fuel production building with a release of the magnitude projected is considered to be beyond extremely unlikely.

The degree of damage incurred and any subsequent release of radioactive materials depends on the size and speed of the aircraft involved. MAR identified in the above scenarios is assumed to be involved and vulnerable to release due to failure of the inert gloveboxes and confinement and subsequent fires. MAR for the feedstock and finished product is assumed stored in an area that is not affected by the crash. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. MAR includes 5,000 grams of plutonium metal that oxidizes after the loss of confinement and inert atmosphere (Section D.3.1.4). MAR contains 4,500 grams of molten plutonium metal released from a casting operations fire (Section D.3.1.5) and 4,500 grams of plutonium from the 3013-container overpressure (Section D.3.1.5). It also includes 398 grams of plutonium from a TRU waste drum fire (Section D.3.1.6), plus 1,000 grams from resin, 246 grams from liquid, and 3,500 grams from an electrorefiner explosion in aqueous/electrorefining fuel preparation (Section D.3.1.7). Deposition during transport through the building or the accident debris is not assumed, so an LPF of 1 is modeled. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

Table D–2. Accident Scenarios and Source Terms for Idaho National Laboratory and K Area VTR Fuel Production Operations

Accident Scenario	*MAR Grams of Pu or Other*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF*	*RF*	*LPF*	*Source Term Dose Equivalent Scale Factor*	*Source Term Pu-239 Dose Equivalent (FP, grams Pu-239)*
D.3.1.1 Criticality while alloying the three components of the metal fuel, uranium, plutonium, and zirconium	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44	Extremely Unlikely	1	1.0	1	1	NA	See Table D-44
D.3.1.2 Fire Impingement on Fuel Material (Intact confinement)	5,000	KIS-grade Pu, ~ 5 kg Pu per assembly	Extremely Unlikely	1	3.0×10^{-5}	0.04	0.1	9.83	5.90×10^{-3}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	5,000	KIS-grade Pu, ~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	1	3.0×10^{-5}	0.04	1	9.83	5.90×10^{-2}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	4,500	5,090 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	1	5.0×10^{-4}	0.5	1	9.83	1.11×10^{1}
D.3.1.5 Plutonium Oxide-to-Metal Conversion– Explosion of 3013 Container of PuO_2	4,500	5,090 g KIS grade PuO_2 in one 3013 container	Extremely Unlikely	1	2.2×10^{-2}	1	0.1	9.83	97.3
D.3.1.6 Beyond-Design-Basis Fire Involving TRU Waste Drum	398	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	1	5.0×10^{-4}	1	1	9.83	1.96

Accident Scenario	MAR Grams of Pu or Other	Assumed Material Type	Frequency (per year)	DR	ARF	RF	LPF	Source Term Dose Equivalent Scale Factor	Source Term Pu-239 Dose Equivalent (FP, grams Pu-239)
D.3.1.7 Aqueous/ Electrorefining Fuel Preparation	1,000 resin 246 nitrate solution 3,500 electrorefiner	KIS-grade Pu	Extremely Unlikely	0.1 resin 1.0 liquid 1 electro-refiner	9×10^{-3} resin 6×10^{-3} liquid 1×10^{-3} electro-refiner	1 1 0.4	0.005	9.83	1.86×10^{-1}
D.3.1.8 Aircraft Crash into VTR Fuel Production Facility	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	varies	varies	varies	1	9.83	1.02×10^{3}
D.3.1.9 Beyond-Design-Basis Earthquake and Fire Involving All of VTR Fuel Production MAR	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	varies	varies	varies	1	9.83	1.02×10^{3}

ARF = airborne release fraction; DR = damage ratio; FP = fission products; g = grams; INL= Idaho National Laboratory; kg = kilograms; KIS = K Area Interim Surveillance; LPF = leak path factor; MAR = material at risk; NA = not applicable; Pu = plutonium; PuO_2 = plutonium oxide; RF = respirable fraction; TRU = transuranic; VTR = Versatile Test Reactor.

D.3.1.9 Beyond-Design-Basis Earthquake and Fire Involving All of Versatile Test Reactor Fuel Production Material at Risk

A beyond-design-basis earthquake is assumed to occur and involve all of the VTR fuel production activities. The MAR identified in the above scenarios is vulnerable to release because of confinement failure of the inert gloveboxes and subsequent fires. The feedstock and finished product are assumed to have a secondary effect on the source term because of the form of the material and are not included in the event. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. MAR includes the same material identified for the aircraft crash scenario (Section D.1.3.8). Deposition during transport through the building or the accident debris is not assumed, so an LPF of 1 is modeled. An earthquake that results in this much damage would require accelerations substantially higher than the design-basis requirements for the structure and major failures of the building and equipment would be expected. This event would be characterized as beyond extremely unlikely. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario

D.3.2 Transuranic Waste Handling Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

TRU waste is waste that is contaminated with alpha emitting TRU radionuclides (radionuclides with atomic numbers greater than that of uranium) that have half-lives greater than 20 years and in which the TRU radionuclide concentrations are greater than 100 nanocuries per gram of waste. TRU waste consists of tools, rags, protective clothing, and other materials contaminated with radioactive elements, such as plutonium. TRU waste would be generated mostly from fuel production and spent fuel treatment. VTR operations and post-irradiation examination of test assemblies would generate less TRU waste than fuel production and spent fuel treatment.

D.3.2.1 Fire Outside Confinement (Waste from Fuel Production or Spent Fuel Treatment)

In this scenario, a fire is postulated to occur in a 1.2 × 1.2 × 2.4-meter (4 × 4 × 8-foot) solid TRU waste box because of spontaneous combustion, pyrophoric material, vehicle accident, electrical failure, or poor housekeeping. The accident is assumed to occur outdoors during handling. The release occurs at ground level over 10 minutes. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The assumed accident frequency is extremely unlikely. MAR is assumed to be one box of TRU waste. MAR is based on one spent fuel pin, with the nuclide distribution of a spent fuel assembly. The material is decayed for a period of four years. A fuel pin is a small item that is representative of activities in either the fuel production or spent fuel treatment facilities. The fuel pin is assumed to represent a bounding quantity of material and is considered to represent the material that would be a contaminant on the items in a TRU waste box. Even though spent fuel has some burnup and contains fission products and activation products, the material still contains a bounding quantity of material for waste from fuel production. The material in a TRU waste box would potentially consist of items, such as contaminated tools, personal protective equipment, and materials (for example rags and blotter paper) used for decontamination activities. The DR is assumed to be 0.001, based on the design of the TRU waste box. Failure of the waste box is assumed to occur because of a seal breach, crack, or puncture and not a massive rupture. The bounding ARF and RF value of 1 is assumed for noble gases. For the other radionuclides, the bounding ARF and RF values of 5×10^{-4} and 1, respectively are based on thermal stresses to packaged surface contaminated waste from DOE-HDBK-3010-94 (DOE 1994). Since the fire is outside building confinement, the LPF is 1. **Table D–3** summarizes MAR, release fractions, and source term for this accident scenario.

Table D–3. Accident Scenarios and Source Terms for the Transuranic Waste Handling Accidents at Savannah River Site, Idaho National Laboratory and Oak Ridge National Laboratory

Accident Scenario	*MAR Grams of Pu or Other*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF*	*RF*	*LPF*	*Source Term Dose Equivalent Scale Factor*	*Source Term No. of Assemblies or grams Pu-239*
D.3.2.1 Fire Outside Confinement (Waste from fuel production or spent fuel treatment)	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	0.001	Noble Gas: 1 Other: 5×10^{-4}	1	1	1.00	4.61×10^{-6}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	3,346 grams	Pu-239 equivalent	Unlikely	0.5	6.0×10^{-3}	0.01	1	1.00	1.00×10^{-1}

Am = americium; ARF = airborne release fraction; DR = damage ratio; INL= Idaho National Laboratory; LPF = leak path factor; MAR = material at risk; ORNL = Oak Ridge National Laboratory; Pu = plutonium; RF = respirable fraction; SRS = Savannah River Site.

D.3.2.2 Fire Outside Involving a Waste Drum with 23 Grams of Americium-241

TRU waste containing americium-241 is generated by fuel production operations. Production waste is placed in containers and temporarily stored (or staged) pending shipment to an offsite disposal facility. A fire is postulated to occur in a 55-gallon drum because of poor housekeeping. The accident is assumed to occur outside confinement during handling. The release occurs at ground level over 10 minutes. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The assumed accident frequency is unlikely. Based on the Waste Isolation Pilot Plant (WIPP) Acceptance Criteria, remote-handled waste is limited to 55-gallon drums with a maximum allowed load of 80 curies or about 23 grams of americium-241. To comply with the WIPP criteria, MAR is assumed to be 1 drum of TRU waste containing 23 grams of americium-241 or 3,346 grams of plutonium-239 equivalent. The DR is assumed to be 0.5 because only a portion of the waste in the container is assumed to be involved in the fire. The bounding ARF and RF values of 6×10^{-3} and 0.01, respectively, are based on thermal stresses to composite solids from DOE-HDBK-3010-94 (DOE 1994). Since the fire is outside building confinement, the LPF is 1. Involved workers could inhale some radioactive material before evacuating the area. The effects from the fire outside confinement bound the effects from a corresponding fire inside confinement or a drop of the drum either inside confinement or outside confinement. Table D–3 summarizes MAR, release fractions, and source term for this accident scenario

D.3.3 Versatile Test Reactor Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

The selection of accidents for the VTR at either INL or ORNL considers various sources of background information. Information in Sections D.3.3.1 through D.3.3.4 is used to select accidents that provide representative radiation and hazardous material effects to receptors associated with siting the VTR at either INL or ORNL. The analysis for these accidents is presented in Section D.3.3.5.

D.3.3.1 Review of Sodium-Cooled Reactor Accidents and Operations

Sodium-cooled reactors have been operated for a number of years. Analysis of sodium-cooled reactor accidents provides insight into possible accident initiators as discussed below (Cahalan 2008).

Experimental Breeder Reactor-1 (EBR-I) – During a test to investigate the prompt positive component of the power coefficient, an unanticipated power excursion resulted in fuel melting. Subsequent investigations identified fuel rod bowing as the source of the positive reactivity feedback.

Fermi-1 – Investigations revealed fuel melting in two adjacent assemblies. Another adjacent assembly was bent, with no internal damage. A crumpled zirconium plate was found in the inlet plenum. Flowing coolant apparently caused the loose zirconium plate to partially or completely cover the inlet nozzle of various assemblies during the multiple start-ups.

Phenix – Four rapid, large, negative reactivity excursions triggered automatic scrams due to power reduction. Given the amplitude and speed of the events, only core movements could cause the observed behavior. The final explanation attributed the reactivity excursions to outward (radial) expansion of the assembly lattice followed by a return to the original position.

Super Phenix – Leaking sodium was contained by the storage tanks' guard vessel. Investigation showed cracks at support plates and tank walls' weld beads. Cracking was associated with the steel material in conjunction with microcracking in zones of high hardness, residual stresses close to the elastic limit of the material, and the hydrogen embrittlement.

MONJU – A sodium leak was detected at a thermocouple well. The thermocouple well tip failed due to flow-induced cycle fatigue.

Fast Flux Test Facility (FFTF) – Analysis of observations during the FFTF acceptance/startup testing program (Wootan et al. 2017) reveals items that may be pertinent to the VTR.

1. Periodic flow and pressure oscillations occurred in the secondary main heat transport system loops. The oscillations were caused by periodic vortex formation and release from piping tees near the inlets of the dump heat exchangers.
2. Fuel cladding breaches can release radioactive cesium (cesium-134 and cesium-137) to the coolant and become a significant source of radiological dose in the plant. Cesium can be effectively removed from the sodium using cesium traps. Because of the relatively high vapor pressure of cesium at reactor operating temperatures, cesium is likely to transport to and throughout the reactor cover's gas system.
3. Gas entrainment and accumulation in the primary coolant could result in a positive reactivity insertion in the reactor if a significant volume of gas were to pass through the core. A similar reactivity insertion could occur in the VTR.
4. FFTF experienced a significant increase in the primary system's sodium pressure drop during its early operation. Pressure normally decreases between inlet and outlet (pressure drop) but the pressure decrease was greater than expected. Observers believed the increased pressure drop was caused by the deposition of silicon-based crystals in the fuel assembly's inlet orifices.
5. Under low flow conditions, sodium can stratify, resulting in large vertical thermal gradients and the potential for undesirable thermal stresses.
6. An inert gas, typically argon, is used as a cover gas over the sodium in the reactor vessel and other components in liquid metal reactors. Due to the high sodium temperature, the cover gas may become saturated with sodium vapor and contain significant quantities of sodium aerosol.

EBR-II – A wide variety of mild undercooling and overpower tests were safely conducted at EBR-II. Identification and investigation of major safety and availability issues surrounding sodium-cooled fast reactor operation with breached fuel elements was conducted through a vigorous run-beyond-cladding-breach testing program. Safety-related experimental programs and modifications in EBR-II resulted in the development and defense of a safety philosophy and documentation to support operation of sodium-cooled fast reactors and development of the first set of technical specifications for a fast reactor power plant in the United States. Additionally operating history and demonstration tests, such as those performed at EBR-II, demonstrated the ability for metal-fueled sodium reactors to be designed in such a manner that core reactivity decreases as temperature increases. This feature would reduce reactor fission power to levels matching heat rejection rates.

A series of tests, originally intended to qualify EBR-II for continued operation, evolved into an experimental program supporting the safety and design of advanced liquid metal reactors. Testing ranged from demonstration of removal of decay heat by natural circulation to whole-plant simulation of unprotected (without scram) loss-of flow (ULOF) and loss-of-heat-sink (ULOHS) accidents from full power and flow. The ULOF and ULOHS tests demonstrated the ability of pool-type, metal-fueled liquid metal reactors to provide self-protection in beyond-design-basis accidents.

D.3.3.2 Current Versatile Test Reactor Safety Basis

The VTR is being designed with the concept of ensuring safety throughout proposed operating conditions, as well as being resilient under potential accident or upset conditions. Consistent with DOE guidance for safety in design, the focus of design is to reduce or eliminate hazards, with a bias towards preventive, as opposed to mitigative, design features and a preference for passive over active safety systems. This general approach creates a design, which is reliable, resilient to upset, and has low potential consequences of accidents.

Safe operation of the VTR is ensured by reliable systems design to ensure preservation of the key reactor safety functions. These key safety functions can be summarized as (1) reactivity control, (2) fission- and decay-heat removal, (3) protection of engineered fission product boundaries, and (4) shielding.

Various systems, which function separately and independently, provide reactivity control in the VTR. The first element of reactivity control is the design of a fuel and core system so that the core experiences a negative reactivity feedback when core temperatures increase. This provides stability in the reactor by ensuring that the power in the reactor cannot "run away" and that deliberate actions are necessary to increase the level of power in the reactor. In addition to this fundamental design expectation, the VTR is designed to be operated without automatic control rod removal. This means that evolutions to increase reactor power are initiated by the operator and are not controlled by an automatic system. This design eliminates automatic rod withdrawal as a potential upset condition. In the event that a plant upset or accident condition results in the need for shutting the reactor down, the reactor can be shut down either through the normal control system slow cutback function or through the scram function. The slow cutback function would drive control rods into the reactor's core until the condition clears or the plant is shutdown. The scram function would shut down the reactor through a VTR Plant Protection System actuation. This system causes an immediate delatching of the control rod drivelines from the control rod drive motors and allows for a spring-assisted gravitational insertion of the control rods into the core. The weight of the control rod and driveline are sufficient to insert the control rod into the core. However, the use of the supporting spring decreases the response time and minimizes the potential for any binding or sticking of control rods. The Plant Protection System, from sensing through logic, and into actuation is an extremely reliable system with independence and diversity in each layer. For additional safety, the VTR core is designed so that the engineered reactivity feedbacks passively reduce reactor power and match residual heat rejection capability simply as a function of the core response to temperatures elevated above the operating temperature, but below cladding failure temperatures (BEA 2020).

Adequate fission product and decay-heat cooling are provided in the design of the VTR through a number of design decisions that ensure robust and resilient mechanisms for heat removal from the reactor and supporting systems. The first design decision is the choice of metal fuel with a sodium bond. The excellent heat transfer capabilities of the metal fuel with sodium bond result in steady state fuel and coolant operating temperature differences, which are much lower than other fuels that have gas gaps and lower heat conduction properties. Use of metal fuel with sodium bond allows greater temperature margins between the fuel temperature and the potential failure temperatures of critical components. In the event that the active mode of the Heat Rejection System fails, the system is designed for each of the loops to have the capacity to reject sufficient heat to prevent fuel damage while functioning in an entirely passive mode. Thus, redundant systems exist that ensure appropriate heat removal capacity. In the event that common issues or very low probability events compromise the ability for both loops of the heat rejection system to reject heat, the ultimate heat removal system, the Reactor Vessel Auxiliary Cooling System (RVACS), would reject the heat. This system allows the heat from the reactor vessel to radiate outward to the collector cylinder. Heat from the collector cylinder and reactor guard vessel would be removed by convection as cold air from outside runs down the cold leg of the RVACS inlet and is heated. The heated air is then exhausted out the RVACS's hot leg stack. One benefit of both the passive mode of the heat removal system and of the RVACS system is that both mechanisms function in a passive state relying only on temperature differences to reject heat. Thus, they are not susceptible to increased temperatures because higher temperatures would result in elevated heat rejection through the dominant heat removal mechanisms of convection and radiation behaviors, respectively (BEA 2020).

Integrity of the barriers designed to provide retention of fission and activation products would be ensured by keeping temperatures and pressures below design limits. A number of barriers would provide mitigation to ensure control of radioactive materials. First, the metal fuel matrix and sodium compatibility would minimize transportation of fission products into the fuel gas gap. Then, the fuel cladding would provide a leak-tight boundary that would contain radioactive materials within the fuel. In the event that some event causes failure of the fuel cladding, fission products would be held up in the primary sodium coolant, the reactor vessel and cover gas cleanup systems, the facility confinement boundary and then, if necessary, the experiment hall confinement boundary and filtration systems (BEA 2020).

A thorough evaluation of the potential abnormal conditions and associated accidents are evaluated in the facility's safety basis documents. The safety analysis of the VTR conceptual design (INL 2019) evaluates accidents involving loss of offsite power, loss of flow, transient overpower, and loss of heat sink with and without the Reactor Protection System (RPS) functioning as designed and demonstrates that fuel failure temperatures would not be exceeded.

D.3.3.3 Key Conclusions from Versatile Test Reactor Safety Analyses

Deterministic safety analyses for the current state of the VTR conceptual design (INL 2019) are being evaluated to support design decisions and analyses that will be required for the future preliminary safety analysis report. The full spectrum of event initiators and subsequent accident sequences would be defined through a probabilistic risk assessment (PRA) of the plant design following the Licensing Modernization Project guidelines. At the current stage of design, transient simulation results for the VTR indicate that large safety margins exist for many event initiators. However, several "enabling" assumptions have been made, in terms of both design features and design limits, in order to perform the analyses; these assumptions may need to be revised as the design matures. Additional events may be evaluated in the future as the VTR design evolves.

Analyses indicate that the offsite source terms for VTR accident scenarios tend to be small because of the melting temperature of the fuel, retention of fission products within the sodium pool (Bucknor et al. 2017), and the small driving force from temperature and pressure in the VTR for release from the confinement to the environment. Transient scenarios for the VTR are presented below.

D.3.3.3.1 Protected Transient Scenarios

The protected transient overpower (PTOP) scenario evaluated in *Safety Analysis for the Versatile Test Reactor Conceptual Design* (ECAR-4733) (INL 2019) is initiated by an unintentional withdrawal of the most reactive control rod. The RPS responds to off-normal conditions by initiating a full system shutdown. The PTOP scenario is a bounding event for perturbations of the reactor core through reactivity changes. The protected-loss-of-heat-sink (PLOHS) accident evaluated in ECAR-4733 is initiated by a simultaneous trip of the secondary electromagnetic (EM) pumps, which significantly reduces heat rejection through the internal heat exchangers (IHXs).

Complete loss of sodium-to-air heat exchanger (SAHX) heat rejection demonstrates how the system responds with the RVACS providing the only source of heat rejection before the RPS detects elevated system conditions. Once elevated temperatures are detected, the RPS responds. The PLOHS is a bounding event for perturbations of the reactor core through core inlet temperature changes.

The protected-loss-of-flow (PLOF)/protected-station-blackout (PSBO) transient evaluated in ECAR-4733 is initiated by an assumed loss of electrical power to all plant systems. Forced circulation in the primary and secondary loops is lost, and heat rejection through the SAHXs is assumed to decrease to zero. The RPS responds to off-normal conditions by initiating a full system shutdown. When the primary pumps trip, core flow decreases to 60 percent and coasts down with a 12-second initial flow halving time thereafter. In the secondary loops, pump head is assumed to decrease from 100 to 0 percent in 1 second. The PSBO is a bounding event for perturbations of the reactor core through mass flow rate changes.

A loss of offsite power (LOOP) considers interruptions of normal power to the electrical buses, which would result in failure of the primary and intermediate EM pumps and reactor scram. Plant-centered LOOP data includes failures of primary safety-class alternating current buses in the plant, which would result in the need to start diesel generators. The LOOP event sequence assumes that all EM pumps are tripped with a LOOP event.

As evaluated in ECAR-4733 (INL 2019), immediately following a seismically-induced LOOP and shutdown of four EM pumps, a scram signal is generated to initiate control rod insertion, and control rods successfully insert. This transient scenario assumes that a reactor shutdown is initiated after detection of a seismic P wave. An assumed five seconds after the shutdown sequence is initiated, the slower, stronger S wave reaches the plant and is assumed to disable the primary pump coast-down mechanisms. Temperatures in the core increase, but because power is significantly reduced and the power-to-flow ratio is less than before the transient began, in-core temperatures do not increase back to the pre-transient values. Even with the coast-down mechanisms being disabled only five seconds after pump trip, the VTR would be able maintain low temperatures and successfully transition to natural circulation.

The PTOP transient described above is analyzed in ECAR-4733 with the addition of an assumed "stuck rod" failure to simulate a malfunction of equipment. That is, in addition to the withdrawal of the most reactive control rod, one of the remaining control or safety rods fails to insert when the RPS initiates a scram. With one of the five control rods that did not withdraw sticking during the RPS-induced scram, the negative reactivity introduced by the scram is reduced. The maximum temperatures predicted during the transients are the same whether or not one rod sticks.

A single coast-down failure during a PTOP or PLOHS transient would not impact peak temperatures because the coast-down failure occurs as part of the transient mitigation by the RPS, that is the reactor is decreasing in power due to insertion of control rods and subsequently, the coast down is being employed. During a PSBO transient, a coast-down failure would occur as part of the transient initiator, so the impact would be more significant. The pump without a functional coast down provides an open flow path back to the cold pool, so some flow would be diverted from the core and instead flow back through the failed pump. Flow in the core would drop quickly and the RPS would initiate a reactor shutdown. All EM pumps and SAHXs would have already tripped, so the only remaining action for the RPS to take would be to scram the control rods. This timing would be the same as for the standard PSBO. The system then would transition to natural circulation and decay heat would slowly decrease.

The review of experiments indicates potential sequences that can lead to an experiment malfunction and a positive reactivity insertion. These include:

1. Insertion from an instrumented assembly
2. Insertion from a non-instrumented assembly
3. Insertion from a test loop
4. Insertion from a rabbit insertion tube
5. Instrumentation or gas line failures that can lead to material discharge into the core

Bounding estimates of the potential design-basis accident reactivity insertion for the above sequences are likely comparable to the transient overpower scenarios analyzed.

D.3.3.3.2 Unprotected Transient Scenarios

A preliminary set of criteria has been developed for evaluating the VTR's response to the unprotected transient scenarios analyzed below. The three unprotected transient scenarios are assessed against criteria roughly based on the PRISM Preliminary Safety Information Document criteria for extremely unlikely events.

The primary considerations for extremely unlikely events are that coolable fuel pin geometry must be maintained and radioactive materials contained. Both can be ensured if the multiple layers of physical protection (i.e., the cladding, reactor vessel, guard vessel, and confinement) remain intact and damage to the reactor does not occur. In these events, a limited amount of fuel melting may be tolerable if the cladding remains intact, and the coolant boundary is not at risk of failure.

The unprotected-transient-overpower (UTOP) scenario evaluated in ECAR-4733 is initiated by an unintentional withdrawal of the most reactive control rod. The UTOP is a bounding event for perturbations of the reactor core through reactivity changes without any mitigating actions by the RPS or reactor operators.

The unprotected-loss-of-heat-sink (ULOHS) transient evaluated in ECAR-4733 is assumed to be initiated by a simultaneous trip of all secondary EM pumps, significantly reducing heat rejection through the IHXs. Additionally, heat rejection through the SAHXs is assumed to be lost. Assuming zero SAHX heat rejection is conservative because, even if the dampers for all ten SAHXs were completely closed, there would still be parasitic heat losses. As with the other unprotected transients, the RPS is assumed to fail to take any action in response to the off-normal conditions. Reactor operator action is also neglected. Therefore, control rods do not scram, and the primary pumps do not trip during the ULOHS. The primary pumps remaining on is the main difference between the ULOHS transient and the unprotected-station-blackout (USBO) transient analyzed below.

The ULOF/USBO transient evaluated in ECAR-4733 is initiated by an assumed loss of electrical power to all plant systems. Forced circulation in the primary and secondary loops is lost, and heat rejection through the SAHXs is assumed to decrease to zero. This transient is initiated by the same failures as the PSBO transient, except that the RPS is assumed to fail to take any action in response to the off-normal conditions. Therefore, control rods would not be scrammed during the USBO. The USBO is a bounding event for perturbations of the reactor core through mass flow rate changes without any mitigating actions by the RPS or reactor operators. The station blackout (SBO) is considered more severe than a loss of flow (LOF) because it also includes a loss-of-heat-sink (LOHS) component due to the simultaneous tripping of the secondary pumps and the loss of heat rejection through the SAHXs. The RVACS is responsible for long-term heat rejection.

D.3.3.4 Versatile Test Reactor Defense-in-Depth

The VTR safety design approach implements the defense-in-depth strategy by adopting the traditional five layers of defense-in-depth to the VTR as follows:

Layer 1: Prevention of abnormal operation and failures

Layer 2: Control of abnormal operation and detection of failures

Layer 3: Control of accidents within the design-basis

Layer 4: Control of severe facility conditions

Layer 5: Mitigation of radiological consequences.

D.3.3.4.1 Prevention of Abnormal Operation and Failures

At the first level, the VTR is designed to operate with a high level of reliability, so that accident initiators are prevented from occurring. The conceptual VTR plant takes advantage of well-established fast reactor metal fuel, sodium coolant, and structural materials that are stable and compatible. VTR design considers the proposed operational ranges for systems and components and ensures that material selection provides for reliable operations during normal operations. Thermal and operational duty cycles are considered in this analysis to ensure that fatigue failures and aging degradations are both understood and minimized. Plant control systems (PCS) are designed as high reliability industrial systems to keep high plant availability. Additionally, adequate instrumentation to ensure reliable plant control and early recognition of abnormal conditions is provided. PCSs are designed with the intent of ensuring that stable plant states are maintained during plant control changes and control variables are measured and evaluated to ensure that changes resulting in abnormal operations are minimal.

D.3.3.4.2 Control of Abnormal Operation and Detection of Failures

Plant instrument and control systems provide the initial protection for ensuring that abnormal operations and deviations are minimized and that conditions are appropriately corrected when failures occur. The PCS contains provisions to detect and provide alarms associated with equipment failures. Actions initiated by the PCS could, as appropriate, decrease reactor power by inserting control rods or throttle primary and secondary EM pumps to ensure heat balance.

In addition, the RPS would initiate a reactor scram, shut off the primary EM pumps, and initiate confinement isolation if identified setpoints were exceeded. Reliability of ensuring the ability of core flow and the transition from forced to natural circulation would be provided by the EM pump coast-down machines, which facilitate applying appropriate control in events where core flow is impacted.

In addition to these active systems, the VTR design benefits from strong favorable reactivity feedbacks that together with the low-pressure sodium coolant and reference metallic fuel, provide passive shutdown and passive safety behavior under various reactor upset conditions.

D.3.3.4.3 Control of Accidents within the Design Basis

This layer of defense-in-depth for the VTR is achieved by conservative design and engineered safety systems for reactor shutdown, decay heat removal, and confinement of radioactive materials. Success in meeting the objectives in this layer is shown by virtue of the fact that all safety basis events are well within the acceptable range, and all design-basis accidents analyzed are successfully mitigated by the safety-class structures, systems, and components performing their intended safety functions. Successive, multiple physical barriers are in place for protection against release of radioactivity and hazardous materials. Multiple, diverse, and independent means are provided to accomplish safety functions. Reactor shutdown systems, actuated by the RPS, and shutdown heat-removal systems provide high reliability protection functions. The selection of liquid sodium coolant and metallic fuel with a pool-type primary system arrangement provides a highly reliable reactor system with large operational safety margins. The coolant thermophysical properties provide superior heat removal and transport characteristics at low operating pressure with a large temperature margin to boiling. The metallic fuel operates at a relatively low temperature, below the coolant boiling point, due to its high thermal conductivity. The pool-type primary system confines all significantly radioactive materials within a single vessel and allows for shutdown heat removal by natural circulation within the vessel and through the RVACS.

D.3.3.4.4 Control of Severe Facility Conditions

This layer's objective is to control severe plant conditions and mitigate beyond extremely unlikely event consequences. The “unprotected” TOP (UTOP), LOHS (ULOHS), and SBO (USBO) transients are initiated by the same sequences of events defined for the protected transients, except the RPS is assumed to fail to take any action. The inherent and passive features of the system are responsible for bringing the system to a stable state at safe temperatures. The UTOP, ULOHS, and USBO scenarios are bounding events for perturbations of the reactor core through reactivity changes, core inlet temperature changes, and core mass flow rates changes, respectively, without any mitigating actions by the RPS or reactor operators. The passive performance mechanisms for ensuring reactivity control and cooling provide performance with generally stronger feedbacks as temperatures increase. These design features help control the level of severity of facility upsets. Additionally, the various levels of confinement barriers (cladding, coolant, reactor vessel, confinement, and experiment hall structure) provide thresholds that serve to control the release of radioactive material if facility conditions are severe enough to result in fuel failures and releases.

D.3.3.4.5 Mitigation of Radiological Consequences

The fifth layer of defense applies to severe accidents where significant releases of radiological or hazardous material could occur. High accident doses to workers and the public could result from severe accidents. However, DOE requirements for emergency planning and the generally large distances to site boundaries at DOE sites, as well as additional safety management programs, provide the ability to mitigate consequences from these extremely low probability events.

D.3.3.4.6 VTR Facility Seismic Hazard Classification

The VTR would satisfy the applicable requirements and criteria contained in DOE-STD-1020-2016. Based on being a hazard category 1 facility, the VTR and support safety-class systems are categorized as SDC-5, and the experiment hall and other facility handling systems are categorized as SDC-3 or less, per the criteria in DOE-STD-1020-2016 (DOE 2016). Such facilities would have to be designed or evaluated for an SDC-3 design-basis earthquake with a mean annual exceedance probability of 1×10^{-4}, corresponding to a return period of 10,000 years or an SDC-5 design-basis earthquake with a mean annual exceedance probability of 1×10^{-5}, corresponding to a return period of 100,000 years.

D.3.3.5 Analysis of Versatile Test Reactor Reactor-Building Accidents

The analysis of reactor building accidents considers accidents that have a low frequency of occurrence, but large consequences and a spectrum of other accidents that have higher frequencies of occurrence and smaller consequences. As indicated in Sections D.3.3.2 and D.3.3.3, most VTR-specific accidents are either prevented or mitigated such that there would be no release of radioactive materials to the environment. An early PRA for the VTR identified a few accident sequences that, at very low probabilities, had the potential for fuel damage and confinement damage. These sequences were predominately initiated by severe seismic events substantially above the seismic design-basis. These beyond extremely unlikely, high-consequence accident sequences presented the highest public radiological risks (probability × consequences) because most of the traditional accidents (for example, loss of coolant accidents) were either prevented or mitigated and did not result in radiological releases.

In addition to the VTR-specific design-basis accidents identified with no radiological releases, accidents resulting in releases from the VTR and the adjoining experiment hall require consideration in NEPA documents. In the VTR building, one possible pathway to the environment would be from releases from the cover gas. These releases would, by design be minuscule. Accidents in the experiment hall involving fuel handling are also a possibility. The ex-vessel hazards and accidents have been determined to involve a mishandling or malfunction of fresh or spent fuel during fuel movements or storage outside the reactor vessel. The VTR may also be vulnerable to beyond-design-basis events, such as severe seismic events, that might fail reactor cooling and confinement. Each of these categories of accidents is addressed in the following sections.

In the analysis presented in the following subsections for VTR-building accidents, the source term for the spent fuel that has 6 percent burnup with no decay time is found in *Probabilistic Risk Assessment, Preliminary Mechanistic Source Term Analysis* (ECAR-4737) (Grabaskas 2019) and is shown in Table D–42 (Attachment D1). The source term for the fresh fuel is calculated with the isotopic compositions shown in the *Evaluation of the VTR Ex-Vessel Inhalation Dose Consequences* (INL 2020b). The source term for the fresh fuel includes uranium at 5 percent enrichment and the isotopic distribution shown in **Table D–4**. (Note that americium is added to account for a 30-year plutonium-241 decay.) **Table D–5** details the pin and assembly mass in grams.

Table D–4. VTR Fresh Assembly

Isotope	*Weight Percent*	*Grams*
Plutonium-238	0.10%	8.9
Plutonium-239	68.70%	6112.2
Plutonium-240	26.40%	2348.8
Plutonium-241	3.40%	75.6
Plutonium-242	1.40%	124.6
Americium-241	75% of 241	226.9
Uranium-234		11.1
Uranium-235	5%	1770.0
Uranium-238		33589.9

VTR = Versatile Test Reactor.

Table D–5. VTR Pin and Assembly Mass

204	grams per fuel pin
41	grams plutonium per pin
217	pins per assembly
8,897	grams plutonium per assembly
35,371	grams uranium per assembly

VTR = Versatile Test Reactor.

D.3.3.5.1 Versatile Test Reactor Design-Basis-Accidents with Releases

The safety analyses for the VTR based on the conceptual design (INL 2019, 2020c), as well as the preliminary *Versatile Test Reactor Probabilistic Risk Assessment Summary* (VTR PRA) (GEH 2019) indicate that most of the accident sequences do not lead to a release of radionuclides from the reactor. The potential of a radioactive material release in the cover gas can be an important factor for operational safety. Leaks may result in the release of radioactive gases from the primary system, including fission gases from failed fuel and activated sodium vapor/aerosols (sodium-23 and argon-41), into the reactor room, other reactor facilities, or the environment. While the cover gas system is at low pressure, portions of it will likely be at relatively high temperature (approximately the temperature of the reactor cover gas within the vessel). Although the system is at low pressure with relatively benign contents consisting mostly of noble gases, the length of piping and number of valves associated with the system provides potential avenues for system leaks.

Releases from the cover gas cleanup system are assessed as part of two different scenarios because the system extends from the reactor head in the reactor room to a service room outside the reactor room. The first scenario examines a major leak of the cover gas cleanup system within the reactor room. A leak in this location would result in the complete release of the contents of the reactor cover gas. However, parts of the system outside the reactor room would be isolated by system check valves. The second scenario assesses a leak from the cover gas cleanup system outside the reactor room, which is where the cesium traps and noble-gas-removal components are located. Half of the cesium inventory is located within the cesium traps and all of the noble gases from the pins are within the noble gas retention components. With a system leak, release of all of the cesium and noble gases within these components is conservatively assumed. In addition, no retention within the service room is assumed, representing an immediate release to the environment with no time for decay. The parts of the cover gas cleanup system within the reactor room are assumed to isolate from the section of the system in the service room.

Cover Gas System Failure with Intact Confinement

The accident is assumed to occur because of an equipment failure in the cover gas system. The assumed accident frequency is unlikely. MAR is assumed to be the gaseous inventory from three fuel pins and activation products in the cover gas. The DR is 0.5 for cesium and 1 for all other isotopes. For generation of vapors plus release from physical confinement, the recommended ARF is 1.0 (DOE 1994). All materials in the gaseous state can be transported and inhaled; therefore, an RF of 1.0 is recommended (DOE 1994). Reduction in the source term because of the reactor confinement is modeled, and an LPF of 6×10^{-5} is used. **Table D–6** summarizes MAR, release fractions, and source term for this accident scenario (BEA 2020).

Table D–6. Source Term for Cover Gas Cleanup System Leak in the Reactor Room

	Material at Risk		*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
	Isotope	*Curies*					
Activation Gases	Ne-23[a]	0	1	1	1	6×10^{-5}	0
	Ar-41	1.12×10^{-3}	1	1	1	6×10^{-5}	6.72×10^{-8}
	Na-22	1.18×10^{-4}	1	1	1	6×10^{-5}	7.08×10^{-9}
	Na-24	1.43×10^{-3}	1	1	1	6×10^{-5}	8.58×10^{-8}
Failed Fuel Pins	Cs-134	3.04×10^{1}	0.5	1	1	6×10^{-5}	9.12×10^{-4}
	Cs-135	1.59×10^{-3}	0.5	1	1	6×10^{-5}	4.77×10^{-8}
	Cs-136	1.09×10^{2}	0.5	1	1	6×10^{-5}	3.27×10^{-3}
	Cs-137	1.06×10^{2}	0.5	1	1	6×10^{-5}	3.18×10^{-3}
	Cs-138	1.13×10^{-4}	0.5	1	1	6×10^{-5}	3.39×10^{-9}
	Kr-83m	2.67	1	1	1	6×10^{-5}	1.60×10^{-4}
	Kr-85	1.62×10^{1}	1	1	1	6×10^{-5}	9.72×10^{-4}
	Kr-85m	1.12×10^{3}	1	1	1	6×10^{-5}	6.72×10^{-2}
	Kr-87	4.44×10^{-5}	1	1	1	6×10^{-5}	2.66×10^{-9}
	Kr-88	5.96×10^{1}	1	1	1	6×10^{-5}	3.58×10^{-3}
	Xe-131m	2.64×10^{3}	1	1	1	6×10^{-5}	1.58×10^{-1}
	Xe-133	4.00×10^{3}	1	1	1	6×10^{-5}	2.40×10^{-1}
	Xe-133m	5.18×10^{3}	1	1	1	6×10^{-5}	3.11×10^{-1}
	Xe-135	5.57×10^{3}	1	1	1	6×10^{-5}	3.34×10^{-1}
	Xe-135m	4.91×10^{3}	1	1	1	6×10^{-5}	2.95×10^{-1}
	Xe-138	3.87×10^{-5}	1	1	1	6×10^{-5}	2.32×10^{-9}

Ar = argon; Cs = cesium; Kr = krypton; Na = sodium; Ne = neon; Xe = xenon.

[a] Ignored due to short half-life (~37 seconds).

Source: BEA 2020.

Release from Cover Gas System Failure with Failed Confinement

The accident is assumed to occur when an earthquake causes a failure of the cover gas piping. The frequency of the earthquake is extremely unlikely. For purposes of this scenario, MAR is assumed as the gaseous inventory from 10 fuel pins even though a fuel failure rate of this magnitude would be a significant operational concern and would likely prompt reactor outages to understand the issues. The DR is 0.5 for cesium and 1 for all other isotopes. For generation of vapors plus release from physical confinement, the recommended ARF is 1.0 (DOE 1994). All materials in the gaseous state can be transported and inhaled; therefore, an RF of 1.0 is recommended (DOE 1994). There is no reduction in the source term because of reactor confinement, and an LPF of 1 is used. **Table D–7** summarizes MAR, release fractions, and source term for this accident scenario (BEA 2020).

Table D–7. Source Term for Cover Gas Cleanup System Leak Outside the Reactor Room

	Material at Risk		*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
	Isotope	*Curies*					
Noble Gas Retention Components	Kr-85	2.90×10^{-1}	1	1	1	1	2.90×10^{-1}
	Kr-87	1.10×10^{-5}	1	1	1	1	1.10×10^{-5}
	Kr-88	3.60×10^{-5}	1	1	1	1	3.60×10^{-5}
	Xe-133	8.60×10^{-3}	1	1	1	1	8.60×10^{-3}
	Xe-135	7.30×10^{-4}	1	1	1	1	7.30×10^{-4}
Cesium Traps	Cs-134	2.00×10^{-1}	0.5	1	1	1	1.00×10^{-1}
	Cs-136	1.88×10^{-3}	0.5	1	1	1	9.40×10^{-4}
	Cs-137	12.2	0.5	1	1	1	6.10
	Cs-138	1.14×10^{-4}	0.5	1	1	1	5.70×10^{-5}

Cs = cesium; Kr = krypton; Xe = xenon.
Source: BEA 2020.

D.3.3.5.2 Versatile Test Reactor Experiment Hall Accidents

A range of operations involving irradiated fuel assemblies and irradiated test assemblies would occur in the experiment hall. These operations could expose the fuel to a range of hazards, including fires and drops due to mechanical failures or human error. These could be initiated by a range of events, including operational accidents and a seismic event. The experiment hall would be designed and evaluated for an SDC-3 design-basis earthquake with a mean annual exceedance probability of 4×10^{-4}, corresponding to a return period of 2,500 years. Thus, seismic events in the extremely unlikely or lower frequency category could possibly be expected to initiate accidents. Four accidents are postulated, including a beyond-design-basis seismic event so severe that collapse of the experiment hall would be possible.

D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire

A fire involving a test assembly in the experiment hall is postulated. The frequency of the seismically-induced fire is extremely unlikely. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. MAR is based on one spent fuel pin, with the nuclide distribution of a spent fuel assembly decayed for 220 days. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For plutonium, americium and other actinides in MAR, the bounding ARF and RF values of 3×10^{-5} and 0.04, respectively (DOE 1994). For uranium, activation products, and fission products in MAR, the bounding ARF and RF values of 1×10^{-3} and 1, respectively (DOE 1994). There is no reduction in the source term because of reactor confinement, so an LPF of 1 is used. **Table D–10** summarizes MAR, release fractions, and source term for this accident scenario.

D.3.3.5.2.2 Fire Involving Versatile Test Reactor Spent Fuel Assemblies

A fire involving spent fuel in the experiment hall is postulated. DOE considers releases from fires involving fresh fuel or releases from fires involving a test assembly to be bounded by fires involving spent fuel. A room fire is assumed to occur and impinge upon a safety-class spent fuel cask designed to withstand the thermal stress from a fire. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The frequency of the imitating fire is unlikely, but a fire of the magnitude postulated coupled with failure of a safety-class cask and heating the spent fuel to eutectic temperatures (800 degrees Celsius (1,470 degrees Fahrenheit) is considered extremely unlikely to beyond extremely unlikely. MAR is assumed as the amount of fuel in 3 spent fuel assemblies that have cooled for 220 days. The DR is 1. The fire event's release fractions are taken from a fuel assembly at eutectic[5] pin failure, but the fuel does not create a self-sustained oxidizing reaction. The bounding ARF and RF values by isotope group are from **Table D–8**. There is no reduction in the source term from the experiment hall confinement, and an LPF of 1 is modeled. Table D–10 summarizes MAR, release fractions, and source term for this accident scenario.

Table D–8. Release Conditions for the Eutectic Spent Fuel Fire in the VTR Experiment Hall

Isotope Group	*Isotope Group ARF × RF*	*Material, Release Conditions, and Reference*
Noble Gases Group: (iodine, bromine)	0.67	The eutectic release fractions are taken from 800 degrees Celsius (1,472 degrees Fahrenheit) eutectic pin failure event. The fractions are the migration fractions interpolated to 6 percent burnup. The resulting fractions may be conservative for a fire involving metal fuel that does not create a self-sustained oxidizing reaction outside of the reactor vessel because it assumes the estimated releases are entirely airborne and respirable.
Alkali Metals Group: (cesium, rubidium)	0.1	
Tellurium Group: (tellurium, antimony, selenium)	0.42	
Barium, strontium Group: (barium, strontium)	6.0×10^{-3}	
Noble Metals Group: (ruthenium, rhodium, palladium, molybdenum, technetium, cobalt)	0.24	
	6.0×10^{-4}	
Cerium Group: (cerium, plutonium, neptunium)	6.0×10^{-4}	
Lanthanides Group: (lanthanum, zirconium, neodymium, niobium, promethium, samarium, yttrium, curium, americium)	6.0×10^{-5}	
Europium Group: (europium)	0.42	
Remaining elements	1.0×10^{-3}	

ARF = airborne release fraction; RF = respirable fraction; VTR = Versatile Test Reactor.
Source: ECAR 4777 Rev. 0, page 3 (INL 2020b).

D.3.3.5.2.3 Versatile Test Reactor Fuel Assembly Drop in Experiment Hall

A drop of spent fuel in the experiment hall is postulated. Drops of fresh fuel are considered to have no release fractions. A drop of spent fuel bounds releases from a drop of a test assembly. The release fractions for the spent fuel drop are listed in **Table D–9**.

[5] The term eutectic refers to a mixture of materials that has a melting point that is lower than the melting points of the separate materials. The term eutectic is important in the fire because the VTR reactor fuel may have a lower melting temperature and larger release fractions than fuel from reactors other than the VTR.

Table D–9. Release Factors for the Spent Fuel Drop in the VTR Experiment Hall

Isotope Group	*Isotope Group ARF × RF*	*Material, Release Conditions, and Reference*
Drop of spent fuel Cold gap Release		A cold gap release occurs when spent fuel element are mechanically damaged, but the temperature is low enough that the cladding does not suffer any thermal damage.
Noble Gases Group:(xenon, krypton)	0.4	
Halogens Group: (iodine, bromine)	0.003	
Alkali Metals Group: (cesium, rubidium)	0.003	
Tellurium Group: (tellurium, antimony, selenium)	1.0×10^{-4}	
Barium, Strontium Group: (barium, strontium)	6.0×10^{-7}	
Noble Metals Group: (ruthenium, rhodium, palladium, molybdenum, technetium, cobalt)	6.0×10^{-7}	
Cerium Group: (cerium, plutonium, neptunium)	6.0×10^{-7}	
Lanthanides Group: (lanthanum, zirconium, neodymium, niobium, promethium, samarium, yttrium, curium, americium)	6.0×10^{-7}	

ARF = airborne release fraction; RF = respirable fraction; VTR = Versatile Test Reactor.
Source: ECAR 4777 Rev. 0, page 4 (INL 2020b).

Given the need for irradiated fuel to be in a cask and the robustness of the casks, the frequency of the drop is assumed to be beyond extremely unlikely. MAR is assumed to be the amount of fuel in 1 spent fuel assembly that has cooled for 220 days. The DR is assumed to be 1. The bounding ARF and RF values are from Table D–9. For isotopes not listed in Table D–9, an ARF of 4×10^{-5} and an RF of 1 for aerodynamic entrainment and resuspension of surface contaminated waste is used (DOE 1994). Reduction in the source term due to experiment hall confinement is modeled (an LPF of 0.005 is used). Table D–10 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.3.5.2.4 Versatile Test Reactor Seismic Event Resulting in Collapse of the Experiment Hall

One accident scenario considers the consequences of a severe seismic event that would collapse the VTR experiment hall. Releases based upon the collapse of the experiment hall would necessarily be limited to the fresh and spent fuels. These fuels would be on their way into or out of the reactor. Given the robust structural protections provided by the Reactor Head Access area floor and other SDC-3 components, the VTR design would meet the satisfaction of the three key safety functions in a passive mode under NPH conditions. This scenario assumes 12 fuel assemblies stored in casks and 6 fuel assemblies in the process of being unloaded or washed.

The release occurs at ground level over 10 minutes. The assumed seismic accident frequency is beyond extremely unlikely. MAR is the equivalent of 18 spent fuel assemblies with the inventory decayed for 220 days. The DR is 0.1, because the building collapse would not be expected to cause extensive damage to spent fuel in casks or in the cleaning station. The ARF and RF for noble gases are 1. For the remaining radionuclides in MAR, an ARF of 1×10^{-3} with an RF of 1.0 is assessed to bound the suspension of (set the limits of) surface contamination from non-brittle solid material for this phenomenon (DOE 1994). The LPF is 1. Confinement in the experiment hall is not modeled. Depending on the location of the drop, nearby workers could inhale the airborne radioactive materials before evacuating the area. Table D–10 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.4 Spent Fuel Handling and Treatment Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

The selection of accidents for spent fuel handling and treatment at either INL or ORNL considers various sources of background information. Because spent fuel treatment involving assemblies or fuel pins occurs in a hot cell filled with inert gas, no releases occur unless the hot cell confinement is breached. Release of fission product gases is anticipated as part of the facility design due to the release that occurs when the spent fuel is melted. The analysis for these accidents is presented in Sections D.3.4.1 through D.3.4.2.

Table D–10. Accident Scenarios and Source Terms for Versatile Test Reactor Operations at Idaho National Laboratory/Materials and Fuels Complex and Oak Ridge National Laboratory

Accident Scenario	*MAR*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term Number of Assemblies*
VTR Operational Accidents								
D.3.3.5.1.1 Cover Gas System Failure with Intact Confinement	3 fuel pins	Gases See Table D–6.	Unlikely	0.5 for Cs, 1 for other gases	1.0	1	6×10^{-5}	Not Applicable
D.3.3.5.1.2 Cover Gas System Failure with Failed Confinement	10 fuel pins	See Table D–7.	Extremely Unlikely	0.5 for Cs, 1 for other gases	1.0	1	1	Not Applicable
Spent Fuel Accidents in the VTR Experiment Hall								
D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire in Experiment Hall	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	220-day cooled spent fuel assembly See Table D-42.	Extremely Unlikely	1	Noble gases: 1; Pu, Am, and other Actinides: 3×10^{-5} U, Activation Products, Fission Products: 1×10^{-3}	0.04 for Pu, Am, and other Actinides 1 for remaining	1	4.61×10^{-3}
D.3.3.5.2.2 Fire at Eutectic Temperature Involving VTR Fuel Assemblies	3 assemblies	220-day cooled spent fuel assembly See Table D-42.	Extremely Unlikely to Beyond Extremely Unlikely	1	Noble Gases: 0.67 Halogens: 0.1 Cs/Rb: 0.42 Te/Sb: 6×10^{-3} Ba/Sr: 0.24 Noble Metals: 6×10^{-4} Ce/Pu: 6×10^{-4} Lanth: 6×10^{-5} Eu: 0.42 Rest 1×10^{-3}	1	1	3
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	1 assembly	220-day cooled spent fuel assembly See Table D-42.	Beyond Extremely Unlikely	1	Noble Gas: 0.4 Halogens: 0.003 Cs/Rb: 0.003 Te/Sb: 1×10^{-4} Ba/Sr: 6×10^{-7} Noble Metals: 6×10^{-7} Ce/Pu: 6×10^{-7} Lanth: 6×10^{-7} Rest: 4×10^{-5}	1	0.005	1

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	DR	ARF [a]	RF	LPF	Source Term Number of Assemblies
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	12 assemblies in casks and 6 assemblies being unloaded or washed. Total MAR 18	220-day cooled spent fuel assembly See Table D-42.	Beyond Extremely Unlikely	0.1	Noble Gas: 1 Halogens: 1×10^{-3} Cs/Rb: 1×10^{-3} Te/Sb: 1×10^{-3} Ba/Sr: 1×10^{-3} Noble Metals: 1×10^{-3} Ce/Pu: 1×10^{-3} Lanth: 1×10^{-3} Eu: 1×10^{-3}	1	1	1.8

Am = americium; ARF = airborne release fraction; Ba = barium; Ce = cerium; Cs = cesium; DR = damage ratio; Eu = europium; HRS = heat removal system; KIS = K Area Interim Surveillance; Lanth = lanthanum; LPF = leak path factor; MAR = material at risk; Pu = plutonium; Rb = rubidium; RF = respirable fraction; RVACS = Reactor Vessel Auxiliary Cooling System; Sb = antimony; Sr = strontium; U = uranium; VTR = Versatile Test Reactor.

[a] Isotope groups are defined in Tables D-8 and D–9.

D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)

A scenario is postulated to examine the consequences of an inadvertent nuclear criticality while spent nuclear fuel is being treated. The criticality results from a seismic event that also causes failure of confinement. The source term for this criticality is based on a fission yield of 1.0×10^{19} fissions. This source term, which is used for all facilities, is based on that given in DOE-HDBK-3010-94 (DOE 1994). Criticality safety controls should prevent this accident from occurring and the materials involved should remain at less than a critical mass. Thus, this accident is identified as extremely unlikely. The scenario represents a metal criticality. A 100 percent release of fission products generated in the criticality is assumed. However, the scenario did not postulate aerosolized, respirable metal fragments to be released. Engineered and administrative controls should be available to ensure that the double-contingency principles are in place for all portions of the process. For purposes of this VTR EIS, the DR is 1, and the bounding ARF and RF value of 1 is assumed for noble gases. Deposition during transport through the building or debris is not assumed, so an LPF of 1 is modeled. **Table D–11** summarizes MAR, release fractions, and source term for this accident scenario.

Table D–11. Accident Scenarios and Terms for Spent Fuel Handling and Treatment Activities at Oak Ridge National Laboratory and Idaho National Laboratory/Materials and Fuels Complex

Accident Scenario	*MAR grams of Pu or other*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term Number of Ingots*
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44	Extremely Unlikely	1	1.0	1	1	See Table D-44
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	1 molten metal spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1	Noble Gas: 1 other: 2×10^{-5}	1	1	1
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	1 spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1	Noble Gas: 1 Pu, Am, and other Actinides: 3×10^{-5} U, Activation Products, Fission Products: 1×10^{-3}	0.04 for Pu, Am, and other Actinides 1 for remaining	1	1

Am = americium; ARF = airborne release fraction; DR = damage ratio; LPF = leak path factor; MAR = material at risk; Pu = plutonium; RF = respirable fraction U = uranium.

[a] Isotope groups are defined in Table D–8 and Table D–9.

D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure

A molten fuel spill into the spent fuel treatment hot cell is assumed to occur when an earthquake causes failure of the hot cell confinement. The frequency of the earthquake and event is extremely unlikely. The hot cell enclosure and the offgas exhaust ventilation system would be expected to fail and allow the release of radioactive material released in the spill. MAR is the amount of fuel in one fuel assembly. The fuel is postulated to have a decay time of 4 years. The DR is assumed to be 1. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For the remaining radionuclides in MAR, the bounding ARF and RF values of 2.0×10^{-5} and 1, respectively are based on a free-fall spill of aqueous solutions (<3 m) with a density >1.2 g/cm^3 from DOE-HDBK-3010-94 (DOE 1994). There is no reduction in the source term because of hot cell confinement; therefore, an LPF of 1 is used. Table D–11 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure

The fire selected for analysis would affect the sodium and fuel material in one spent fuel assembly. The event is assumed to occur when an earthquake causes confinement failure in conjunction with the drop of an assembly that breaches the assembly cladding. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The frequency of the sodium fire is extremely unlikely. MAR is assumed to be the amount of fuel in one assembly. The fuel is postulated to have a decay time of 4 years, and the DR is 1. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For the plutonium, americium, and other actinides in MAR, the bounding ARF and RF values are 3×10^{-5} and 0.04, respectively (DOE 1994). For uranium, activation products, and fission products in MAR, the bounding ARF and RF values are 1×10^{-3} and 1, respectively (DOE 1994). There is no reduction in the source term based on confinement, and an LPF of 1 is used. Table D–11 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.5 Post-Irradiation Examination Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

The selection of accidents for post-irradiation examination at either INL or ORNL considers various sources of background information. Because post-irradiation examination involving assemblies or fuel pins occurs in a hot cell filled with inert gas, no releases occur unless the hot cell confinement is breached. Release of fission product gases is anticipated as part of the facility design due to the release that occurs when the experiment is prepared for further examination.

D.3.5.1 Fire Involving Test Assembly (Seismically-Induced Confinement Failure)

The fire selected for analysis impacts the test assembly during post-irradiation examination. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. Because post-irradiation examination is performed in a hot cell with an inert argon atmosphere, a fire would not occur unless the hot cell confinement is breached. For this scenario, an earthquake is postulated to breach the hot cell confinement. The frequency of the earthquake is extremely unlikely. MAR is assumed to be the amount of fuel in one-half of a spent fuel assembly. Selecting one-half of a spent fuel assembly as MAR accounts for multiple VTR experiments being examined concurrently in the hot cell. The fuel is postulated to have a decay time of 4 years, and the DR is 1. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For the remaining radionuclides in MAR, the bounding ARF and RF values of 3×10^{-5} and 0.04, respectively are for airborne release of particulates during complete oxidation of metal mass (DOE 1994). The release factors are specifically for an airborne release of plutonium particulates formed by oxidation at elevated temperatures without self-sustained oxidation (DOE 1994). There is no reduction in the source term from the hot cell confinement, and an LPF of 1 is used. **Table D–12** summarizes MAR, release fractions, and source term for this accident scenario.

Table D–12. Accident Scenarios and Source Terms for Post-Irradiation Examination Activities at Idaho National Laboratory/Materials and Fuels Complex and Oak Ridge National Laboratory

Accident Scenario	*MAR Grams of Pu or Other*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term Number of Assemblies*
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	0.5 assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1	Noble Gas: 1 Other: 3×10^{-5}	0.04	1	0.5

ARF = airborne release fraction; DR = damage ratio; LPF = leak path factor; MAR = material at risk; Pu = plutonium; RF = respirable fraction.

D.3.6 Spent Fuel Storage Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

The selection of accidents for spent fuel storage at either INL or ORNL considers various sources of background information. The analysis for these accidents is presented in Sections D.3.6.1 through D.3.6.3.

D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask

After spent fuel has been washed, it is loaded into spent fuel storage casks and stored on a spent fuel storage pad pending transfer to the fuel treatment facility where it will be treated and conditioned for eventual disposal. Additionally, until a disposal site has been identified, the spent fuel storage casks containing conditioned spent fuel ingots are stored on a spent fuel storage pad. The accident is assumed to occur when an earthquake causes failure of the spent fuel storage casks. The release occurs because of damage to the contents of several spent fuel storage casks. The release occurs at ground level over 10 minutes. The assumed accident frequency is beyond extremely unlikely. MAR is the equivalent of six spent fuel assemblies with an inventory decay time of 4 years, and the DR is 0.5, because damage to material in the cask is not expected to be extensive. For purposes of this VTR EIS, the ARF and RF value of 1 is assumed for noble gases. For the remaining radionuclides in MAR, an ARF of 4×10^{-5} and an RF of 1 for aerodynamic entrainment and resuspension of surface contaminated waste is used (DOE 1994). The LPF is assumed to be 1. The source term is not reduced because of the confinement provided by a damaged spent fuel storage cask. Depending on the location of the drop, nearby workers could inhale the airborne radioactive materials before evacuating the area. **Table D–13** summarizes MAR, release fractions, and source term for this accident scenario.

Table D–13. Accident Scenarios and Source Terms for Spent Fuel Storage Activities at Idaho National Laboratory/Materials and Fuels Complex and Oak Ridge National Laboratory

Accident Scenario	*MAR*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term Number of Assemblies*
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	6 spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43.	Beyond Extremely Unlikely	0.5	Noble Gas: 1 Halogens: 4×10^{-5} Cs/Rb: 4×10^{-5} Te/Sb: 4×10^{-5} Ba/Sr: 4×10^{-5} Noble Metals: 4×10^{-5} Ce/Pu: 4×10^{-5} Lanth: 4×10^{-5} Eu: 4×10^{-5}	1	1	3
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	1.0×10^{19} fissions, Three spent fuel assemblies	Nobles+ Iodine See Table D-44.	Extremely Unlikely	1	1.0	1	1	See Table D-44
D.3.6.3 Drop of Fuel-Loaded Cask	6 spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43.	Extremely Unlikely	0.5	Noble Gas: 1 Halogens: 4×10^{-5} Cs/Rb: 4×10^{-5} Te/Sb: 4×10^{-5} Ba/Sr: 4×10^{-5} Noble Metals: 4×10^{-5} Ce/Pu: 4×10^{-5} Lanth: 4×10^{-5} Eu: 4×10^{-5}	1	0.5	3

ARF = airborne release fraction; Ba = barium; Ce = cerium; Cs = cesium; DR = damage ratio; Eu = europium; Lanth = lanthanum; LPF = leak path factor; MAR = material at risk; Pu = plutonium; Rb = rubidium; RF = respirable fraction; Sb = antimony; Sr = strontium; Te = tellurium.

[a] Isotope groups are defined in Table D–8 and Table D–9.

D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask

An inadvertent nuclear criticality is postulated to occur in untreated fuel from a spent fuel storage cask. The criticality results from a seismic event that also causes failure of the spent fuel storage cask. The source term for this criticality is based on a fission yield of 1.0×10^{19} fissions. This source term, which is used for all facilities, is based on that given in DOE-HDBK-3010-94 (DOE 1994). The scenario represents a metal criticality. The metal is postulated to soften, resulting in a 100 percent release of fission products stored in the fuel and fission products generated in the criticality. The fission products stored in the fuel are from three spent fuel assemblies with a decay time of four years. However, no aerosolized, respirable metal fragments are predicted to be released. This accident is conservatively estimated to have a frequency of extremely unlikely. For purposes of this VTR EIS, the DR is 1, and the bounding ARF and RF value of 1 is assumed for noble gases. Deposition debris from spent fuel storage cask does not reduce the source term, so an LPF of 1 is assumed. Table D–13 summarizes MAR, release fractions, and source term for this accident scenario

D.3.6.3 Drop of Fuel-Loaded Cask

After spent fuel has been washed, it is loaded into spent fuel storage casks pending transfer to the fuel treatment site. The accident is assumed to occur when a spent fuel storage cask is dropped during handling. The cask drop is assumed to be caused by equipment failure or human error. The release occurs from damaging the contents of one spent fuel storage cask. The release occurs at ground level over 10 minutes. The assumed accident frequency is extremely unlikely. MAR is the equivalent of six spent fuel assemblies. The DR is assumed to be 0.5 because not all fuel in the cask is expected to be involved. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For the remaining radionuclides in MAR, an ARF of 4×10^{-5} and an RF of 1 for aerodynamic entrainment and resuspension of surface contaminated waste is used (DOE 1994). The dropped cask is assumed to continue providing some confinement after the drop. Consequently, the LPF is assumed 0.5. Depending on the location of the drop, nearby workers could inhale the airborne radioactive materials before evacuating the area. Table D–13 summarizes MAR, release fractions, and source term for this accident scenario.

D.4 Radiological Impacts of Facility Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

The following sections summarize the consequence of accidents described in Sections D.3.1 through D.3.6. The radiological and hazardous material consequences are presented for accidents at the SRS, INL, and ORNL. Because few sources of energy and only limited materials would be present, the likelihood of a major accident is extremely remote. Most incidents would not involve much energy. Any criticality, fire, drop, or spill would be confined, with little or no radiological impact. For bounding accidents, radiological impacts on workers in the immediate vicinity of the incident and on those exposed to released material could be relatively high. The radiological impacts from beyond-design-basis earthquakes on involved and noninvolved workers could be high as well.

For most of the proposed activities, accidents that could potentially affect noninvolved workers, the maximally exposed offsite individual, and the public are extremely unlikely or beyond extremely unlikely, i.e., an annual probability of one in 10,000 or less. Accidents with releases of this magnitude are not expected to occur during the lifetime of the project. Impacts on the noninvolved worker are modeled assuming that worker is 330 feet downwind of the release point. Dispersion modeling of impacts so close to a release point are highly dependent on assumptions and does not include protection of buildings and other structures, turbulence, personnel taking emergency actions, etc. As a result, these models are, therefore, highly uncertain. They do offer an opportunity to compare the relative impacts of each accident but are much less useful for comparing impacts at different sites because any real differences due to meteorology or location would be small compared to the uncertainties in modeling close-in effects.

The potential near-term impacts from the initial plume passage are reported as the "Near-Term-Dose" in the following tables while the long-term impacts of exposure to the radionuclides after the plume passage are added to the "Near-Term-Dose" and reported as the "Near+Long-Term Dose." The long term (or chronic) dose includes the combined effects of ingesting contaminated foods, direct radiation exposure from residual material on the ground (ground shine), inhalation of disturbed, residual ground-level particulates (resuspension), and ingestion of contaminated water.

D.4.1 Radiological Impacts from Accidents at Idaho National Laboratory and Oak Ridge National Laboratory Versatile Test Reactor-Related Facilities

Table D–14 through **Table D–19** summarize the impacts related to various accident scenarios for reactor fuel production, TRU waste handling, VTR operation, spent fuel handling, post-irradiation examination, and spent fuel storage at MFC.

D.4.1.1 Accident Impacts from VTR Fuel Production Capability at Idaho National Laboratory

Two phases of reactor fuel production are evaluated. The first phase is a feedstock preparation capability. This phase involves the receipt of plutonium that contains high levels of impurities, e.g., americium-241 (which is present as the result of the decay of plutonium-241 in "older" plutonium) or plutonium in other than metal form (e.g., plutonium oxide). The feedstock preparation capability would address the issues of plutonium polishing and conversion to metal. The second phase of reactor fuel production would involve the alloying of plutonium with uranium and zirconium, and fabricating fuel pins and fuel assemblies.

Table D–14 presents the potential impacts from accidents associated with implementation of VTR fuel production capability at INL. For the VTR fuel fabrication activities at either the INL Site or SRS, the bounding operational accident is a fire that results in heating and over pressurization of a 3013 can of plutonium oxide. Releases from other accidents such as a fire and spill involving molten uranium and plutonium while being cast into fuel would be filtered before release to the environment. During all of the VTR fuel fabrication operations, the radiological materials are either in metal form in a container designed to contain the radionuclides in virtually all accidents or in an inert glovebox. As such, only controlled, filtered releases would be expected. Workers would be protected from routine accidents within the gloveboxes and the ventilation system that directs glovebox releases through HEPA filters and to an outside stack.

For accidents associated with the INL VTR metal preparation portion of the fuel fabrication option, the bounding accident from plutonium "polishing" operations would be an uncontrolled reaction during portions of the aqueous operations. This could release radioactive materials to the glovebox, but any releases to from the glovebox or room to the environment would be filtered and have low impacts. If plutonium oxide were used as a feed material, the bounding operational event is a high-pressure rupture of a welded, DOE standard 3013, plutonium-oxide storage container. This could occur if the container were exposed to a fire that burns sufficiently long to raise the internal pressure of the container to the point of rupture. This scenario, while theoretically possible, would be extremely unlikely and normal fire prevention and mitigation practices should reduce the chance of it occurring.

In a severe seismic event, the building and glovebox structures could be damaged enough that an unfiltered release could occur. Even so, most of the material not specifically being processed at the time of the earthquake would be in metal form and would not result in a substantial release even with loss of glovebox and building integrity.

A beyond-design-basis earthquake was also postulated that could threaten unfiltered releases of all of the MAR in the Fuel Preparation and Fabrication area, including the molten plutonium in casting, liquid plutonium in the polishing operation, and plutonium oxide in the conversion operations. In a severe seismic event, the building and glovebox structures could be damaged enough that an unfiltered release could occur. Even so, most of the material not specifically being processed at the time of the earthquake would be in metal form and would not result in a substantial release even with loss of glovebox and building integrity.

D.4.1.2 Accident Impacts from VTR Transuranic Waste Activities at Idaho National Laboratory

Table D–15 presents the potential impacts from accidents associated with implementation of VTR fuel production capability at INL. The bounding accident is a fire outside of a building's confinement system involving americium-241 waste. The waste container would be designed to prevent such fires.

D.4.1.3 Accident Impacts from Reactor Operations at Idaho National Laboratory

Table D–16 presents the potential impacts from accidents associated with operation of the VTR at INL. Both reactor-specific and VTR building accidents are considered. The potential impacts of a hypothetical, beyond-design-basis reactor accident with loss of cooling are presented in Section D.4.9.

D.4.1.3.1 Reactor Accidents

As discussed in Section D.3.3.5.1, the results of the safety analyses for the VTR based on the conceptual design (INL 2019, 2020c), as well as the preliminary VTR PRA (GEH 2019) indicate that most of the accident sequences do not lead to a release of radionuclides from the reactor. The only accidents with potential releases at probabilities of 10^{-6} per year or greater are leaks from the cover gas system which lead to minor accidental releases and are not evaluated in detail.

D.4.1.3.2 Reactor Building Fuel Accidents

As discussed in Section D.3.3.5.2, the results of the safety analyses indicate that accidents associated with operations involving irradiated fuel assemblies and irradiated test assemblies in the experiment hall area of the reactor building could occur and bound, in terms of consequences, all reactor-area accidents. As indicated in Table D–16, all of the accidents evaluated involve spent fuel assemblies or test assemblies. Most handling accidents involving these assemblies would not result in substantial releases. The bounding accidents evaluated are assumed to be initiated by a major event such as a major fire or severe, beyond-design-basis seismic event. As indicated in the Table D–16, the largest impacts are associated with a major fire that heats spent fuel assemblies inside a safety-class cask (designed to withstand fires) to the point of failure of the cask. This combination of events is expected to fall in the extremely unlikely to beyond extremely unlikely frequency category since it requires multiple failures of safety systems. The potential radiological impact, are relatively high, with a dose of 0.24 Rem to the MEI and a near-term population dose of 36 person-rem. The fire, assumed to be 800 degrees Celsius (1,470 degrees Fahrenheit), is sufficient high to release relatively high fractions of isotopes of elements such as cesium and strontium that could result in high doses if ingested. Without mitigation measures, the long term dose due primarily to ingestion is 4,300 person-rem or 3 LCFs among the 50-mile population. It is expected that as the design of the VTR fuel handling systems within the experiment hall progresses, designs with ensure that this accident is prevented or mitigated. The other bounding accident is a seismic event that results in collapse of the experiment hall. Because all of the spent fuel would be stored in safety class casks, only spent fuel being handled might be vulnerable. The potential radiological impact, are moderately high, with a dose of 0.071 rem to the MEI and a near-term population dose of 13 person-rem. Without mitigation measures, the long term dose due to resuspension and ingestion is 27 person-rem and no LCFs among the 50-mile population.

D.4.1.4 Accident Impacts from Treatment at Idaho National Laboratory

Table D–17 presents the potential impacts from accidents associated with spent fuel handling and treatment at MFC. Because operations occur in a hot cell designed to contain high quantities of radionuclides, uncontrolled releases to the environment do not occur unless there is a major breach of confinement. Accidental handling and spills within the hot would not result in uncontrolled releases. Each of the accident scenarios selected for further evaluation requires breach of hot cell confinement, likely be a major seismic event. Such an event could lead to ingress of oxygen into the hot cell and sodium fires. The highest impacts are associated with a seismically-initiated hot-cell confinement failure and sodium fire involving a spent fuel assembly and cladding. Such an event is considered extremely unlikely.

D.4.1.5 Accident Impacts from Post-Irradiation Examination at Idaho National Laboratory

Table D–18 presents the potential impacts from accidents associated with post-irradiation examination at MFC. Because operations occur in a hot cell designed to contain high quantities of radionuclides, uncontrolled releases to the environment do not occur unless there is a major breach of confinement. The accident scenarios selected for further evaluation requires breach of hot cell confinement, likely be a major seismic event. Such an event could lead to ingress of oxygen into the hot cell and a fire with a test assembly under examination. Such an event is considered extremely unlikely.

D.4.1.6 Accident Impacts from Spent Fuel Storage at Idaho National Laboratory

Table D–19 presents the potential impacts from accidents associated with spent fuel handling and storage at MFC. VTR spent fuel would be stored in robust canisters designed to withstand a wide-range of accidents. As such, most types of handling accidents, including impacts and handling accidents, would not be expected to result in releases. The accidents identified result in small releases.

D.4.2 Radiological Impacts from Accidents at Versatile Test Reactor-Related Facilities at Oak Ridge National Laboratory

Table D–20 through **Table D–24.** summarize the impacts related to various accident scenarios for TRU waste handling, VTR operation, spent fuel handling, post-irradiation examination, and spent fuel storage at ORNL. Because few sources of energy and only limited materials would be present, the likelihood of a major accident is extremely remote. Most incidents would not involve much energy, and any criticality, fire, drop, or spill would be confined, with little or no radiological impact. For the bounding accidents, radiological impacts on workers in the immediate vicinity of the incident and on those exposed to released material could be relatively high. The radiological impacts from beyond-design-basis earthquakes on noninvolved workers could be high as well.

The accident scenarios for the VTR alternative and supporting operations at ORNL are identical to those presented for the comparable facilities at INL. No new or substantially different accident scenarios were identified with the placement of the facilities at ORNL. While there are differences in the sites that could affect the probabilities of certain scenarios, particularly natural phenomena-initiate events such as wind, flooding, seismic, and volcanism, the dominant accident scenarios remain the same.

There are, however, differences in potential impacts due to population distributions around the VTR location, distances to the nearest potential offsite individual, and differences in meteorological conditions. Impacts on the maximally exposed member of the public, assumed to be a boater on an arm of Melton Lake about 0.5 miles from the VTR complex are higher than those for the INL site, where the MEI is assumed be on the nearby highway about 3.1 miles away. The offsite population density also is quite different between the two VTR sites. Most of the differences in population impacts between the two sites are directly due to a lower 50-mile population at INL and few people residing within 10 miles of the proposed VTR site.

D.4.2.1 Accident Impacts from VTR Transuranic Waste Activities at Oak Ridge National Laboratory

Table D–20 presents the potential impacts from accidents associated with implementation of VTR fuel production capability at ORNL.

D.4.2.2 Accident Impacts from Reactor Operations at Oak Ridge National Laboratory

Table D–21 presents the potential impacts from accidents associated with operation of the VTR at ORNL. The accident scenarios for the VTR alternative and supporting operations at ORNL are identical to those presented for the comparable facilities at INL. No new or substantially different accident scenarios were identified with the placement of the facilities at ORNL. While there are differences in the sites that could affect the probabilities of certain scenarios, particularly natural phenomena-initiate events such as wind, flooding, seismic, and volcanism, the dominant accident scenarios remain the same. The VTR and support facilities would be designed to the same standards at either site. For natural-phenomena events, the standards are based to return intervals or frequency per year. Thus a VTR at ORNL may be designed to a higher seismic loading in order to meet the DOE standards. The likelihood of beyond-design-basis events should be the same for both sites.

D.4.2.3 Accident Impacts from VTR Support Activities at Oak Ridge National Laboratory

Table D–22, D–23, and D–24 presents the potential impacts from accidents associated VTR support activities, including spent fuel handling and treatment, post-irradiation examination, and spent fuel storage at ORNL. The accident scenarios are identical to those identified for INL.

D.4.3 Radiological Impacts from Accidents at Versatile Test Reactor-Related Fuel Production Activities at K-Reactor Complex at Savannah River Site

Table D–25 and **D–26** summarize the impacts related to various accident scenarios for fuel production at SRS. The accident scenarios for the VTR fuel fabrication operations at SRS are identical to those presented for the comparable facilities at INL. No new or substantially different accident scenarios were identified with the placement of the facilities at SRS. While there are differences in the sites that could affect the probabilities of certain scenarios, particularly natural phenomena-initiate events such as wind, flooding, seismic, and volcanism, the dominant accident scenarios remain the same.

There are, however, differences in potential impacts due to population distributions around the VTR location, distances to the nearest potential offsite individual, and differences in meteorological conditions.

Because few sources of energy and only limited materials would be present, the likelihood of a major accident is extremely remote. Most incidents would not involve much energy, and any criticality, fire, drop, or spill would be confined, with little or no radiological impact. For the bounding accidents, radiological impacts on workers in the immediate vicinity of the incident and on those exposed to released material could be relatively high. The radiological impacts from beyond-design-basis earthquakes on involved and noninvolved workers could be high as well. Impacts on the noninvolved worker are included for the receptor at 330 feet and 3,300 feet. The impacts for the receptor at 330 feet are presented for consistency with the other calculations in this VTR EIS, while the impacts for the receptor at 3,300 feet are presented for historical consistency. Previous SRS EIS documents reported calculation results for a noninvolved worker at 3,300 feet.

Table D–14. Accident Impacts for the VTR Fuel Production Capability at Idaho National Laboratory

Accident Scenario	MAR Grams of Pu or Other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.1.1 Criticality while alloying the three components of the metal fuel (uranium, plutonium, and zirconium)	1.0×10^{19} fissions	Nobles+Iodine See Table D-44	Extremely Unlikely	4.8×10^{-2}	3×10^{-5}	3.6×10^{-3}	2×10^{-6}	1.3×10^{-1}	8×10^{-5}	7.7×10^{-1}	5×10^{-4}
D.3.1.2 Fire Impingement on Fuel Material (intact confinement)	5,000	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely	1.6×10^{-5}	1×10^{-8}	6.4×10^{-6}	4×10^{-9}	1.1×10^{-3}	7×10^{-7}	1.6×10^{-3}	9×10^{-7}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	5,000	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	4.8×10^{-2}	3×10^{-5}	6.5×10^{-5}	4×10^{-8}	1.0×10^{-2}	6×10^{-6}	1.5×10^{-2}	9×10^{-6}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	4,500	5,090 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	9.0	5×10^{-3}	1.2×10^{-2}	7×10^{-6}	2.0	1×10^{-3}	2.9	2×10^{-3}
D.3.1.5 Plutonium Oxide-to-Metal Conversion–Explosion of 3013 Container of PuO_2	4,500	5,090 g KIS grade PuO_2 in one 3013 container	Extremely Unlikely	2.7×10^{-1}	2×10^{-4}	1.1×10^{-1}	6×10^{-5}	18	1×10^{-2}	26	2×10^{-2}
D.3.1.6 Beyond-Design-Basis Fire Involving a TRU Waste Drum	398	450 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	1.6	1×10^{-3}	2.2×10^{-3}	1×10^{-6}	0.35	2×10^{-4}	0.51	3×10^{-4}
D.3.1.7 Aqueous/ Electrorefining Fuel Preparation	1,000 resin 246 nitrate solution 3,500 electro-refiner	KIS-grade Pu	Extremely Unlikely	5.2×10^{-4}	3×10^{-7}	2.0×10^{-4}	1×10^{-7}	3.4×10^{-2}	2×10^{-5}	5.0×10^{-2}	3×10^{-5}

Accident Scenario	MAR Grams of Pu or Other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.1.8 Aircraft Crash into VTR Fuel Production Facility	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	8.3×10^{2}	1	1.1	7×10^{-4}	1.8×10^{2}	1×10^{-1}	2.7×10^{2}	2×10^{-1}
D.3.1.9 Beyond-Design-Basis Earthquake and Fire Involving All of VTR Fuel Production MAR	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	8.3×10^{2}	1	1.1	7×10^{-4}	1.8×10^{2}	1×10^{-1}	2.7×10^{2}	2×10^{-1}

g = gram; kg = kilogram; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; Pu = plutonium; PuO_2 = plutonium oxide; rem = roentgen equivalent man; TRU = transuranic; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–2. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Near+Long-Term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–15. Accident Impacts from VTR Transuranic Waste Activities at Idaho National Laboratory

Accident Scenario	MAR Grams of Pu or Other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.2.1 Fire Outside Confinement (waste from fuel production or spent fuel treatment)	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	6.0×10^{-5}	4×10^{-8}	7.4×10^{-8}	4×10^{-11}	1.3×10^{-5}	8×10^{-9}	2.6×10^{-5}	2×10^{-8}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	3,346	Pu-239 Equivalent	Unlikely	8.0×10^{-2}	5×10^{-5}	1.1×10^{-4}	7×10^{-8}	1.8×10^{-2}	1×10^{-5}	1.8×10^{-2}	1×10^{-5}

Am = americium; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man; Pu = plutonium; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–3. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less.

Table D–16. Accident Impacts for VTR Accidents at Idaho National Laboratory/Materials and Fuels Complex

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.3.5.1.1 Cover Gas System Failure with Intact Confinement	3 fuel pins	gases see Table D–7	Unlikely	The source term for this scenario is very small in relation to the source terms for other VTR accidents and no calculation is performed since the results from other calculations are enveloping.							
D.3.3.5.1.2 Cover Gas System Failure with Failed Confinement	10 fuel pins	see Table D–8	Extremely Unlikely	The source term for this scenario is very small in relation to the source terms for other VTR accidents and no calculation is performed since the results from other calculations are enveloping.							
D.3.3.5.2.1 Test Assembly Failure Following Seismically-Induced Fire	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	220-day cooled spent fuel assembly See Table D-42	Extremely Unlikely	5.2×10^{-2}	3×10^{-5}	6.5×10^{-5}	4×10^{-8}	1.2×10^{-2}	7×10^{-6}	3.6×10^{-2}	2×10^{-5}
D.3.3.5.2.2 Release from Fire at Eutectic Temperature Involving VTR Fuel Assemblies	3 assemblies	220-day cooled spent fuel assembly See Table D-42	Extremely Unlikely to Beyond Extremely Unlikely	1.6×10^{2}	2×10^{-1}	2.4×10^{-1}	1×10^{-4}	3.6×10^{1}	2×10^{-2}	4.3×10^{3}	3
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	1 assembly	220-day cooled spent fuel assembly See Table D-42	Beyond Extremely Unlikely	7.3×10^{-4}	4×10^{-7}	1.3×10^{-6}	8×10^{-10}	1.7×10^{-4}	1×10^{-7}	4.6×10^{-2}	3×10^{-5}
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	12 assemblies in casks and 6 assemblies being unloaded or washed. Total MAR 18	220-day cooled spent fuel assembly See Table D-42	Beyond Extremely Unlikely	5.8×10^{1}	7×10^{-2}	7.1×10^{-2}	4×10^{-5}	1.3×10^{1}	8×10^{-3}	2.7×10^{1}	2×10^{-2}

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]

HRS = heat removal system; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man; RVACS = Reactor Vessel Auxiliary Cooling System; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–11. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less.

Table D–17. Accident Impacts for the Spent Fuel Handling and Treatment at Idaho National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44	Extremely Unlikely	1.0	6×10^{-4}	3.9×10^{-3}	2×10^{-6}	1.3×10^{-1}	8×10^{-5}	7.6×10^{-1}	5×10^{-4}
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	1 molten metal spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	6.6×10^{-1}	4×10^{-4}	8.0×10^{-4}	5×10^{-7}	1.4×10^{-1}	9×10^{-5}	2.9×10^{-1}	2×10^{-4}
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	1 spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	9.0	5×10^{-3}	1.1×10^{-2}	7×10^{-6}	2.0	1×10^{-3}	7.8	5×10^{-3}

INL= Idaho National Laboratory; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–12. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–18. Accident Impacts for Post-Irradiation Examination at Idaho National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	One-half of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1.6×10^{-2}	9×10^{-6}	1.9×10^{-5}	1×10^{-8}	3.5×10^{-3}	2×10^{-6}	6.9×10^{-3}	4×10^{-6}

INL= Idaho National Laboratory; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–13. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–19. Accident Impacts for Spent Fuel Storage at Idaho National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	Six spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43	Beyond Extremely Unlikely	3.1	2×10^{-3}	3.9×10^{-3}	2×10^{-6}	6.9×10^{-1}	4×10^{-4}	1.4	8×10^{-4}
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44 4-year cooled spent fuel assembly	Extremely Unlikely	1.0	6×10^{-4}	3.9×10^{-3}	2×10^{-6}	1.3×10^{-1}	8×10^{-5}	7.6×10^{-1}	5×10^{-4}
D.3.6.3 Drop of Fuel-Loaded Cask	Six spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1.6	9×10^{-4}	1.9×10^{-3}	1×10^{-6}	3.5×10^{-1}	2×10^{-4}	6.9×10^{-1}	4×10^{-4}

LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–14. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–20. Accident Impacts from VTR Transuranic Waste Activities at Oak Ridge National Laboratory

Accident Scenario	MAR grams of Pu or other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.2.1 Fire Outside Confinement (Waste from fuel production or spent fuel treatment)	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1.5×10^{-4}	9×10^{-8}	5.4×10^{-6}	3×10^{-9}	5.2×10^{-4}	3×10^{-7}	7.2×10^{-4}	4×10^{-7}
D.3.2.5 Fire Outside Involving a Waste Drum with 23 grams of Am-241	3,346	Pu-239 Equivalent	Unlikely	2.1×10^{-1}	1×10^{-4}	7.2×10^{-3}	4×10^{-6}	0.69	4×10^{-4}	0.69	4×10^{-4}

INL= Idaho National Laboratory; Kg = kilogram; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; Pu = plutonium; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man; Pu = plutonium; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–3. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–21. Accident Impacts for VTR Accidents at Oak Ridge National Laboratory

Accident Scenario	*MAR*	*Assumed Material Type*	*Frequency (per year)*	*Impacts on Noninvolved Worker (330 feet)*		*Impacts on an MEI at 0.5 Miles*		*Near-Term Impacts on Population within 50 Miles*		*Near+Long-Term Impacts on Population within 50 Miles*	
				Dose (rem) [a]	*Probability of an LCF* [b]	*Dose (rem)* [a]	*Probability of an LCF* [b]	*Early Dose (rem)* [a]	*LCFs* [c]	*Early + Chronic Dose (rem)* [a]	*LCFs* [c]
D.3.3.5.1.1 Cover Gas System Failure with Intact Confinement	Three fuel pins	gases see Table D–6	Unlikely	The source term for this scenario is very small in relation to the source terms for other VTR accidents and no calculation is performed since the results from other calculations are enveloping.							
D.3.3.5.1.2 Cover Gas System Failure with Failed Confinement	Ten fuel pins	see Table D–7	Extremely Unlikely	The source term for this scenario is very small in relation to the source terms for other VTR accidents and no calculation is performed since the results from other calculations are enveloping.							
D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire	1 pin or 1/217 (or $4.6{\times}10^{-3}$) of an assembly	220-day cooled spent fuel assembly See Table D-42	Extremely Unlikely	$1.3{\times}10^{-1}$	$8{\times}10^{-5}$	$4.6{\times}10^{-3}$	$3{\times}10^{-6}$	$4.5{\times}10^{-1}$	$3{\times}10^{-4}$	$7.1{\times}10^{-1}$	$4{\times}10^{-4}$
D.3.3.5.2.2 Fire at Eutectic Temperature Involving VTR Fuel Assemblies	Three assemblies	220-day cooled spent fuel assembly See Table D-42	Extremely Unlikely to Beyond Extremely Unlikely	$4.0{\times}10^{2}$	$5{\times}10^{-1}$	1.4	$9{\times}10^{-3}$	1,400	$9{\times}10^{-1}$	15,000	9
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	One assembly	220-day cooled spent fuel assembly See Table D-42	Beyond Extremely Unlikely	$1.9{\times}10^{-3}$	$1{\times}10^{-6}$	$6.7{\times}10^{-5}$	$4{\times}10^{-8}$	$6.7{\times}10^{-3}$	$4{\times}10^{-6}$	$1.3{\times}10^{-1}$	$8{\times}10^{-5}$
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	12 assemblies in casks and 6 assemblies being unloaded or washed. Total MAR 18	220-day cooled spent fuel assembly See Table D-42	Beyond Extremely Unlikely	$1.5{\times}10^{2}$	$2{\times}10^{-1}$	5.1	$3{\times}10^{-3}$	380	$2{\times}10^{-1}$	700	$4{\times}10^{-1}$

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]

HRS = heat removal system; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man; RVACS = Reactor Vessel Auxiliary Cooling System; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–11. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–22. Accident Impacts for the Spent Fuel Handling and Treatment at Oak Ridge National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	1.0×10^{19} fissions	Nobles+Iodine See Table D-44	Extremely Unlikely	2.5	2×10^{-3}	1.5×10^{-1}	9×10^{-5}	4.8	3×10^{-3}	6.2	4×10^{-3}
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	one molten metal spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1.7	1×10^{-3}	5.8×10^{-2}	4×10^{-5}	5.6	3×10^{-3}	7.9	5×10^{-3}
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	one spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	23	3×10^{-2}	8.0×10^{-1}	5×10^{-4}	77	5×10^{-2}	110	7×10^{-2}

LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–12. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–23. Accident Impacts for Post-Irradiation Examination at Oak Ridge National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	one-half of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	4.0×10^{-4}	2×10^{-5}	1.4×10^{-3}	8×10^{-7}	1.3×10^{-1}	8×10^{-5}	1.9×10^{-1}	1×10^{-4}

LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–13. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–24. Accident Impacts for Spent Fuel Storage at Oak Ridge National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	Six spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43	Beyond Extremely Unlikely	8.0	5×10^{-3}	2.8×10^{-1}	2×10^{-4}	27	2×10^{-2}	138	2×10^{-2}
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	1.0×10^{19} fissions Three spent fuel assemblies	Nobles+ Iodine See Table D-44 4-year cooled spent fuel assembly	Extremely Unlikely	2.5	2×10^{-3}	1.5×10^{-1}	9×10^{-5}	4.8	3×10^{-3}	6.2	4×10^{-3}
D.3.6.3 Drop of Fuel-Loaded Cask	Six spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	4.0	2×10^{-3}	1.4×10^{-1}	8×10^{-5}	13	8×10^{-3}	19	1×10^{-2}

LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–14. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–25. Accident Impacts for the VTR Fuel Production Capability at Savannah River Site

Accident Scenario	MAR grams of Pu or other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on Noninvolved Worker (3,300 feet)		Impacts on an MEI at Site Boundary		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.1.1 Criticality while alloying the three components of the metal fuel, uranium, plutonium, and zirconium)	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44	Extremely Unlikely	4.0×10^{-2}	2×10^{-5}	2.4×10^{-2}	1×10^{-5}	1.5×10^{-3}	9×10^{-7}	8.9×10^{-1}	5×10^{-4}	1.7	1×10^{-3}
D.3.1.2 Fire Impingement on Fuel Material (Intact confinement)	5,000	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely	4.4×10^{-6}	3×10^{-9}	4.7×10^{-5}	3×10^{-8}	2.5×10^{-6}	1×10^{-9}	7.0×10^{-3}	4×10^{-6}	9.7×10^{-3}	6×10^{-6}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	5,000	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	4.7×10^{-2}	3×10^{-5}	1.3×10^{-3}	8×10^{-7}	2.5×10^{-5}	1×10^{-8}	6.8×10^{-2}	4×10^{-5}	9.4×10^{-2}	6×10^{-5}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	4,500	5,090 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	8.8	5×10^{-3}	2.4×10^{-1}	1×10^{-4}	4.6×10^{-3}	3×10^{-6}	13	8×10^{-3}	18	1×10^{-2}
D.3.1.5 Plutonium Oxide-to-Metal Conversion	4,500	5,090 g KIS grade PuO_2 in one 3013 container	Extremely Unlikely	7.2×10^{-2}	4×10^{-5}	7.8×10^{-1}	5×10^{-4}	4.1×10^{-2}	2×10^{-5}	120	7×10^{-2}	160	9×10^{-2}

				Impacts on Noninvolved Worker (330 feet)		***Impacts on Noninvolved Worker (3,300 feet)***		***Impacts on an MEI at Site Boundary***		***Near-Term Impacts on Population within 50 Miles***		***Near+Long-Term Impacts on Population within 50 Miles***	
Accident Scenario	***MAR grams of Pu or other***	***Assumed Material Type***	***Frequency (per year)***	***Dose (rem)*** [a]	***Probability of an LCF*** [b]	***Dose (rem)*** [a]	***Probability of an LCF*** [b]	***Dose (rem)*** [a]	***Probability of an LCF*** [b]	***Early Dose (rem)*** [a]	***LCFs*** [c]	***Early + Chronic Dose (rem)*** [a]	***LCFs*** [c]
D.3.1.6 Beyond-Design-Basis Fire Involving a TRU Waste Drum	398	450 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	1.5	9×10^{-4}	4.2×10^{-2}	3×10^{-5}	8.2×10^{-4}	5×10^{-7}	2.2	1×10^{-3}	3.1	2×10^{-3}
D.3.1.7 Aqueous/ Electrorefining Fuel Preparation	1,000 resin 246 nitrate solution 3,500 electrorefiner	KIS-grade Pu	Extremely Unlikely	1.4×10^{-4}	8×10^{-8}	1.5×10^{-3}	9×10^{-7}	7.8×10^{-5}	5×10^{-8}	2.2×10^{-1}	1×10^{-4}	3.0×10^{-1}	2×10^{-4}
D.3.1.8 Aircraft Crash into VTR Fuel Production Facility	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu,	Beyond Extremely Unlikely	8.1×10^{2}	1	22	3×10^{-2}	4.3×10^{-1}	3×10^{-4}	1.2×10^{3}	7×10^{-1}	1,600	1
D.3.1.8 Beyond-Design-Basis Earthquake and Fire Involving All of VTR Fuel Production MAR	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	8.1×10^{2}	1	22	3×10^{-2}	4.3×10^{-1}	3×10^{-4}	1.2×10^{3}	7×10^{-1}	1,600	1

g = grams; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man; SRS = Savannah River Site; Pu = plutonium; PuO_2 = plutonium oxide; TRU = transuranic; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–2. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–26. Accident Impacts from VTR Transuranic Waste Activities at Savannah River Site

Accident Scenario	MAR grams of Pu or other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on Noninvolved Worker (3,300 feet)		Impacts on an MEI at Site Boundary		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.2.1 Beyond-Design-Basis TRU Waste Drum Fire Outside Confinement (Waste from fuel production)	396 (450g PuO_2)	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	7.7×10^{-1}	5×10^{-4}	2.1×10^{-2}	1×10^{-5}	4.1×10^{-4}	2×10^{-7}	1.1	7×10^{-4}	1.5	9×10^{-4}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	3,346	Pu-239 Equivalent	Unlikely	8.0×10^{-2}	5×10^{-5}	2.2×10^{-3}	1×10^{-6}	4.2×10^{-5}	3×10^{-8}	1.2×10^{-1}	7×10^{-5}	1.6×10^{-1}	1×10^{-4}

Am = americium; g = gram; kg = kilogram; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ; Pu = plutonium; PuO_2 = plutonium oxide; rem = roentgen equivalent man; SRS = Savannah River Site; TRU = transuranic; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–3. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

D.4.4 Comparison of the Radiological Impacts from Accidents at Versatile Test Reactor-Related Facilities at Idaho National Laboratory and Oak Ridge National Laboratory

In order to facilitate comparisons of accident impacts for comparable activities at INL and ORNL, the key impacts from Sections D.4.1 and D.4.2 have been extracted to generate tables to enable direct comparisons. Impacts are expressed in terms of the potential LCFs if an accident occurred. Because the noninvolved worker at both sites is 330 feet away from the release point, the differences in modeling impacts should be within the overall uncertainty in meteorology and dispersion. Hence, any differences are not real discriminators and are not reported in these comparison tables. For the MEI, the reported LCF can be interpreted as the probability that the affected individual would ultimately die of an LCF. Unless the exposure is quite high (~1,000 rem), the expected LCF would be 0.

Table D–27 summarizes the impacts related to various accident scenarios for VTR operation, spent fuel handling, post-irradiation examination, TRU waste handling, and spent fuel storage at INL and ORNL. For the VTR at INL or ORNL, the reactor would be designed and operations would be conducted so that for credible accidents, no fuel would melt and accidents would result in negligible releases. A fire involving VTR spent fuel assemblies in the VTR experiment hall is postulated as the bounding operational accident at the VTR. This type of accident is modeled as extremely unlikely to beyond extremely unlikely but may not be credible.

For the VTR support activities, including the post-irradiation examination and spent fuel handling, conditioning, and storage, the bounding operational accidents are a criticality, which results in unfiltered releases, and a filtered release from fire and spills involving molten spent fuel. In a severe seismic event, building structures, process enclosures, and process equipment could be damaged enough that unfiltered releases could occur. This event has lower impacts than a fire involving VTR spent fuel assemblies in the VTR experiment hall. The impacts from the hypothetical beyond-design-basis earthquake at the VTR dominate releases that might occur from all VTR support activities. Results differ between the INL VTR Alternative and the ORNL VTR Alternative because each has unique meteorology, receptor distance, and population distribution.

D.4.5 Comparison of Radiological Impacts from Accidents at Versatile Test Reactor-Related Fuel Production Activities at Idaho National Laboratory and K-Reactor Complex at Savannah River Site

Table D–28 summarizes the impacts related to various accident scenarios for reactor fuel production at INL and SRS. As with Table D–27, this table presents side-by-side results of similar accidents at the two sites. For the VTR fuel production activities at either INL or SRS, the bounding operational accident is a high-pressure explosion of a 3013 container of plutonium oxide Although this event is considered extremely unlikely, it does have the potential for a large release. Releases from other accidents involving liquid, oxide, or molten forms of plutonium would be filtered before release to the environment. In a severe, beyond-design-basis earthquake, severe structural and process equipment damage was postulated. This damage could result in spillage and unfiltered release of liquid, oxide, and molten forms of plutonium. In a severe seismic event, the building and glovebox structures could be damaged enough that an unfiltered release could occur. Even so, most of the material not specifically being processed at the time of the earthquake would be in metal form and would not result in a substantial release even with loss of glovebox and building integrity. Results differ between the INL options and the SRS options because of differences in stack height, meteorology, receptor distance, and population distribution.

Table D–27. Summary of Potential Annual Radiological Impacts for VTR Operational Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

Accident Scenario	*Frequency (per year)*	*MEI (probability of an LCF)*		*Population (Near-Term) (LCFs)* [a,b]		*Population (Near+Long-Term) (LCFs)* [a,b]	
		INL	*ORNL*	*INL*	*ORNL*	*INL*	*ORNL*
VTR Accident Impacts							
D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire in Experiment Hall	Extremely Unlikely	4×10^{-8}	3×10^{-6}	7×10^{-6}	3×10^{-4}	2×10^{-5}	4×10^{-4}
D.3.3.5.2.2 Fire at Eutectic Temperature Involving VTR Fuel Assemblies	Extremely Unlikely to Beyond Extremely Unlikely	1×10^{-4}	9×10^{-3}	2×10^{-2}	9×10^{-1}	3	9
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	Beyond Extremely Unlikely	8×10^{-10}	4×10^{-8}	1×10^{-7}	4×10^{-6}	3×10^{-5}	8×10^{-5}
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	Beyond Extremely Unlikely	4×10^{-5}	3×10^{-3}	8×10^{-3}	3×10^{-1}	2×10^{-2}	4×10^{-1}
Spent Fuel Handling and Treatment							
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	Extremely Unlikely	2×10^{-6}	9×10^{-5}	8×10^{-5}	3×10^{-3}	5×10^{-4}	4×10^{-3}
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	Extremely Unlikely	5×10^{-7}	4×10^{-5}	9×10^{-5}	3×10^{-3}	2×10^{-4}	5×10^{-3}
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	Extremely Unlikely	7×10^{-6}	5×10^{-4}	1×10^{-3}	5×10^{-2}	5×10^{-3}	7×10^{-2}
Transuranic Waste Accident Impacts							
D.3.2.1 Fire Outside Confinement (Waste from fuel INL or spent fuel treatment)"	Extremely Unlikely	4×10^{-11}	3×10^{-9}	8×10^{-9}	3×10^{-7}	2×10^{-8}	4×10^{-7}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	Unlikely	7×10^{-8}	4×10^{-6}	1×10^{-5}	4×10^{-4}	2×10^{-5}	6×10^{-4}
Post-Irradiation Examination Accident Impacts							
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	Extremely Unlikely	1×10^{-8}	8×10^{-7}	2×10^{-6}	8×10^{-5}	4×10^{-6}	1×10^{-4}
Spent Fuel Storage Accident Impacts							
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	Beyond Extremely Unlikely	2×10^{-6}	2×10^{-4}	4×10^{-4}	2×10^{-2}	8×10^{-4}	2×10^{-2}
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	Extremely Unlikely	2×10^{-6}	9×10^{-5}	8×10^{-5}	3×10^{-3}	5×10^{-4}	4×10^{-3}
D.3.6.3 Drop of Fuel-Loaded Cask	Extremely Unlikely	1×10^{-6}	8×10^{-5}	2×10^{-4}	8×10^{-3}	4×10^{-4}	1×10^{-2}

Am = americium; INL= Idaho National Laboratory; LCF = latent cancer fatality; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; VTR = Versatile Test Reactor.

[a] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less. For the MEI, the reported LCF can be interpreted in the probability that the affected individual would ultimately die of a LCF. Unless the exposure is quite high (~1,000 rem), the expected LCF would be 0.

[b] Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage without mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passes. These doses include ingestion of contaminated foods,

Accident Scenario	***Frequency (per year)***	***MEI (probability of an LCF)***		***Population (Near-Term) (LCFs)*** [a,b]		***Population (Near+Long-Term) (LCFs)*** [a,b]	
		INL	***ORNL***	***INL***	***ORNL***	***INL***	***ORNL***

water, etc., direct exposure to deposited material, and resuspension and inhalation of deposited materials. For purposes of the EIS, no interdiction or mitigation is assumed, but such measures would likely occur. The long-term impacts reported include both the near-term and long-term impacts without mitigation.

Table D–28. Summary of Potential Radiological Impacts for Fuel Fabrication Accidents at Idaho National Laboratory and Savannah River Site

Accident Scenario	***Frequency (per year)***	***MEI (probability of an LCF)***		***Population (Near-Term) (LCFs)*** [a, b]		***Population (Near+Long-Term) (LCFs)*** [a, b]	
		INL	***SRS***	***INL***	***SRS***	***INL***	***SRS***
Fuel Fabrication – Accident Impacts							
D.3.1.1 Criticality while alloying the three components of the metal fuel, uranium, plutonium, and zirconium	Extremely Unlikely	2×10^{-6}	9×10^{-7}	8×10^{-5}	5×10^{-4}	5×10^{-4}	1×10^{-3}
D.3.1.2 Fire Impingement on Fuel Material (Intact confinement)	Extremely Unlikely	4×10^{-9}	1×10^{-9}	7×10^{-7}	4×10^{-6}	9×10^{-7}	6×10^{-6}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	Extremely Unlikely to Beyond Extremely Unlikely	4×10^{-8}	1×10^{-8}	6×10^{-6}	4×10^{-5}	9×10^{-6}	6×10^{-5}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	Extremely Unlikely to Beyond Extremely Unlikely	7×10^{-6}	3×10^{-6}	1×10^{-3}	8×10^{-3}	2×10^{-3}	1×10^{-2}
D.3.1.5 Plutonium Oxide-to-Metal Conversion - Explosion of 3013 Container of PuO_2	Extremely Unlikely	6×10^{-5}	2×10^{-5}	1×10^{-2}	7×10^{-2}	2×10^{-2}	1×10^{-1}
D.3.1.6 Beyond-Design-Basis Fire Involving TRU Waste Drum	Extremely Unlikely to Beyond Extremely Unlikely	1×10^{-6}	5×10^{-7}	2×10^{-4}	1×10^{-3}	3×10^{-4}	2×10^{-3}
Fuel Fabrication – Feedstock Preparation – Accident Impacts							
D.3.1.7 Aqueous/Electrorefining Fuel Preparation	Extremely Unlikely	1×10^{-7}	5×10^{-8}	2×10^{-5}	1×10^{-4}	3×10^{-5}	2×10^{-4}
Fuel Fabrication + Feedstock Preparation: Combined Beyond-Design-Basis Earthquake Accident Impacts							
D.3.1.8 Aircraft Crash into VTR Fuel Fabrication Facility	Beyond Extremely Unlikely	7×10^{-4}	3×10^{-4}	1×10^{-1}	7×10^{-1}	2×10^{-1}	1
D.3.1.9 Beyond-Design-Basis Earthquake Involving All VTR Fuel Fabrication and Preparation MAR	Beyond Extremely Unlikely	7×10^{-4}	3×10^{-4}	1×10^{-1}	7×10^{-1}	2×10^{-1}	1

INL= Idaho National Laboratory; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; PuO_2 = plutonium oxide; SRS = Savannah River Site; TRU = transuranic; VTR = Versatile Test Reactor.

[a] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

[b] Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage without mitigation measures, such as sheltering in place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passes. These doses include ingestion of contaminated foods, water, etc., direct exposure to deposited material, and resuspension and inhalation of deposited materials. For

Accident Scenario	Frequency (per year)	MEI (probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	SRS	INL	SRS	INL	SRS

purposes of the EIS, no interdiction or mitigation is assumed, but such measures would likely occur. The long-term impacts reported include both the near-term and long-term impacts without mitigation.

D.4.6 Comparison of the Annual Radiological Risks from Accidents at Versatile Test Reactor-Related Facilities at Idaho National Laboratory and Oak Ridge National Laboratory

Table D–29 summarizes the annual radiological accident risks for specific accident scenarios for VTR operation, spent fuel handling, post-irradiation examination, TRU waste handling, and spent fuel storage at INL and ORNL. Risks are expressed in terms of the product of the estimated annual probability of the accident times the projected impacts, in terms of LCFs, if these accidents were to occur. It is recognized that there are large uncertainties involved in both the probabilities of specific accidents as well as the potential consequences. Nevertheless, expressing the potential impacts in terms of an annual risk to the MEI or general population does provide a useful tool for comparing disparate events, such as severe reactor accidents and simple glovebox accidents. It also allows comparison of events that differ significantly in probability and consequence. For example, the risk to the MEI from both unlikely events with low consequences can be compared to the risks from extremely unlikely events with higher consequences.

For purposes of calculating risk in this EIS, events categorized as “unlikely” are assigned a probability of 10^{-2}. Events categorized as “extremely unlikely” are assigned a probability of 10^{-4}. Events categorized as “extremely unlikely to beyond extremely unlikely” are assigned a probability of 10^{-5}. Events categorized as “beyond extremely unlikely” are assigned a probability of 10^{-6}.

Use of LCF risks allows comparison of individual accidents. For the VTR at INL or ORNL, operational reactor-specific accidents are all managed so that no fuel would melt and releases would be negligible. A fire involving VTR spent fuel assemblies in the VTR experiment hall (D.3.3.5.2.2 Fire Involving VTR Fuel Assemblies) is postulated as the highest risk, operational accident at the VTR. This accident is certainly in the extremely unlikely to beyond extremely unlikely category and may not be credible once the VTR design is completed. Other accidents that contribute substantially to the overall risk are the sodium fire involving spent fuel (D.3.4.3) and the waste drum fire with americium-241 (D.3.2.2). Table D–29 indicates that the overall, near-term LCF risk to the population within 50 miles is 5×10^{-7} at INL and 2×10^{-5} at ORNL.

Some of the accidents release radionuclides that would result in longer-term exposures due to ingestion if no interdiction or mitigation occurred. At both sites, the highest-risk accident for long-term exposure is the fire involving spent fuel assemblies in the experiment hall, which presents the dominant long-term risk to the public. That fire is projected to release substantial quantities of cesium and strontium isotopes that are the dominant contributors to long-term dose.

Also presented at the end of Table D–29 is the annual LCF risk to the MEI and the population within 50 miles. For the MEI, the annual probability of a LCF from VTR reactor and support operations at INL and ORNL is estimated to be 3×10^{-9} and 2×10^{-7}, respectively. The risk is higher at ORNL because the MEI is calculated for a point on Melton Hill Lake within the ORR where the public has access, though by boat. For the population within 50 miles, the annual probability of a LCF from VTR reactor and support operations at INL and ORNL due to near and long-term exposures is estimated to be 3×10^{-5} and 1×10^{-4}, respectively. Here, the primary difference is due to the higher population density near ORNL. The long-term risks to the public from accidents at either site do not take into account mitigation actions DOE would take if an accident occurred.

Table D–29. Summary of Annual Risks for VTR Operational Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

Accident Scenario	Frequency (per year)	MEI (Probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	ORNL	INL	ORNL	INL	ORNL
VTR Accident LCF Risks [c]							
D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire in Experiment Hall	Extremely Unlikely	4×10^{-12}	3×10^{-10}	7×10^{-10}	3×10^{-8}	2×10^{-9}	4×10^{-8}
D.3.3.5.2.2 Fire Involving VTR Fuel Assemblies	Extremely Unlikely to Beyond Extremely Unlikely	1×10^{-9}	9×10^{-8}	2×10^{-7}	9×10^{-6}	3×10^{-5}	9×10^{-5}
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	Beyond Extremely Unlikely	8×10^{-16}	4×10^{-14}	1×10^{-13}	4×10^{-12}	3×10^{-11}	8×10^{-11}
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	Beyond Extremely Unlikely	4×10^{-11}	3×10^{-9}	8×10^{-9}	3×10^{-7}	2×10^{-8}	4×10^{-7}
Spent Fuel Handling and Treatment Accident LCF Risks [c]							
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	Extremely Unlikely	2×10^{-10}	9×10^{-9}	8×10^{-9}	3×10^{-7}	5×10^{-8}	4×10^{-7}
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	Extremely Unlikely	5×10^{-11}	4×10^{-9}	9×10^{-9}	3×10^{-7}	2×10^{-8}	5×10^{-7}
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	Extremely Unlikely	7×10^{-10}	5×10^{-8}	1×10^{-7}	5×10^{-6}	5×10^{-7}	7×10^{-6}
Transuranic Waste Accident LCF Risks [c]							
D.3.2.1 Fire Outside Confinement (Waste from fuel fabrication or spent fuel treatment)"	Extremely Unlikely	4×10^{-15}	3×10^{-13}	8×10^{-13}	3×10^{-11}	2×10^{-12}	4×10^{-11}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	Unlikely	7×10^{-10}	4×10^{-8}	1×10^{-7}	4×10^{-6}	2×10^{-7}	6×10^{-6}
Post-Irradiation Examination Accident LCF Risks [c]							
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	Extremely Unlikely	1×10^{-12}	8×10^{-11}	2×10^{-10}	8×10^{-9}	4×10^{-10}	1×10^{-8}
Spent Fuel Storage Accident LCF Risks [c]							
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	Beyond Extremely Unlikely	2×10^{-12}	2×10^{-10}	4×10^{-10}	2×10^{-8}	8×10^{-10}	2×10^{-8}
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	Extremely Unlikely	2×10^{-10}	9×10^{-9}	8×10^{-9}	3×10^{-7}	5×10^{-8}	4×10^{-7}
D.3.6.3 Drop of Fuel-Loaded Cask	Extremely Unlikely	1×10^{-10}	8×10^{-9}	2×10^{-8}	8×10^{-7}	4×10^{-8}	1×10^{-6}
Total Annual LCF Risk from Reactor and Support Facility Operations without Hypothetical Beyond-Design-Basis Reactor Accident [d]		**3×10^{-9}**	**2×10^{-7}**	**5×10^{-7}**	**2×10^{-5}**	**3×10^{-5}**	**1×10^{-4}**

Accident Scenario	Frequency (per year)	MEI (Probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	ORNL	INL	ORNL	INL	ORNL

Am = americium; INL= Idaho National Laboratory; LCF = latent cancer fatality; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; VTR = Versatile Test Reactor.

[a] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[b] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[c] Risks are the product of the probability and impacts. For purposes of this EIS, events categorized as "unlikely" are assigned a probability of 10^{-2}. Events categorized as "extremely unlikely" are assigned a probability of 10^{-4} Events categorized as "extremely unlikely to beyond extremely unlikely" are assigned a probability of 10^{-5}. Events categorized as "beyond extremely unlikely" are assigned a probability of 10^{-6}.

[d] The total annual risk from reactor and support facility operations is the sum of all accidents evaluated. For the MEI, it represents a reasonable estimate of the annual risk that the MEI might receive sufficient dose to result in a LCF. For the population within 50 miles, it represents a reasonable estimate of the annual number of LCFs that might occur within that population due to accidents.

D.4.7 Comparison of the Annual Radiological Risks from Accidents at Versatile Test Reactor-Related Fuel Production Activities at Idaho National Laboratory and K-Reactor Complex at Savannah River Site

Table D–30 summarizes the annual radiological accident risks for specific accident scenarios for reactor fuel production at INL or SRS. For the VTR fuel production activities at either INL or SRS, the highest risk bounding operational accident is a high-pressure explosion of 3013 container of plutonium oxide. Although this event is considered extremely unlikely, it does have the potential for a large radiological release.

Releases from other accidents involving liquid, oxide, or molten forms of plutonium would be filtered before release to the environment. In a beyond-design-basis earthquake, severe damage to the structures and process equipment is postulated. This could result in spillage and unfiltered release of liquid, oxide, and molten forms of plutonium. In a severe seismic event, the building and glovebox structures could be damaged enough that an unfiltered release could occur. Even so, most of the material not specifically being processed at the time of the earthquake would be in metal form and would not result in a substantial release even with loss of glovebox and building integrity. Results differ between the INL option and the SRS option because of stack height, meteorology, receptor distance, and population distribution.

Table D–30. Summary of Annual Risks for Reactor Fuel Production Accidents at Idaho National Laboratory and Savannah River Site

Accident Scenario	Frequency (per year)	MEI (Probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	SRS	INL	SRS	INL	SRS
Fuel Fabrication – Accident LCF Risks [c]							
D.3.1.1 Criticality while alloying the three components of the metal fuel, uranium, plutonium, and zirconium)	Extremely Unlikely	2×10^{-10}	9×10^{-11}	8×10^{-9}	5×10^{-8}	5×10^{-8}	1×10^{-7}

Accident Scenario	Frequency (per year)	MEI (Probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	SRS	INL	SRS	INL	SRS
D.3.1.2 Fire Impingement on Fuel Material (Intact confinement)	Extremely Unlikely	4×10^{-13}	1×10^{-13}	7×10^{-11}	4×10^{-10}	9×10^{-11}	6×10^{-10}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	Extremely Unlikely to Beyond Extremely Unlikely	4×10^{-13}	1×10^{-13}	6×10^{-11}	4×10^{-10}	9×10^{-11}	6×10^{-10}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	Extremely Unlikely to Beyond Extremely Unlikely	7×10^{-11}	3×10^{-11}	1×10^{-8}	8×10^{-8}	2×10^{-8}	1×10^{-7}
D.3.1.5 Plutonium Oxide-to-Metal Conversion - Explosion of 3013 Container of PuO_2	Extremely Unlikely	6×10^{-9}	2×10^{-9}	1×10^{-6}	7×10^{-6}	2×10^{-6}	1×10^{-5}
D.3.1.6 Beyond-Design-Basis Fire Involving TRU Waste Drum	Extremely Unlikely to Beyond Extremely Unlikely	1×10^{-11}	5×10^{-12}	2×10^{-9}	1×10^{-8}	3×10^{-9}	2×10^{-8}
Feedstock Preparation– Accident LCF Risks [c]							
D.3.1.7 Aqueous/Electro-refining Fuel Preparation	Extremely Unlikely	1×10^{-11}	5×10^{-12}	2×10^{-9}	1×10^{-8}	3×10^{-9}	2×10^{-8}
Fuel Fabrication + Feedstock Preparation: Combined Beyond-Design-Basis Earthquake Accident Risks [c]							
D.3.1.8 Aircraft Crash into VTR Fuel Fabrication Facility	Extremely Unlikely to Beyond Extremely Unlikely	7×10^{-10}	3×10^{-10}	1×10^{-7}	7×10^{-7}	2×10^{-6}	1×10^{-6}
D.3.1.9 Beyond-Design-Basis Earthquake Involving All VTR Fuel Fabrication and Preparation MAR	Beyond Extremely Unlikely	7×10^{-10}	3×10^{-10}	1×10^{-7}	7×10^{-7}	2×10^{-7}	1×10^{-6}
Total Annual LCF Risk from Fuel Fabrication Operations [d]		**8×10^{-9}**	**3×10^{-9}**	**1×10^{-6}**	**8×10^{-6}**	**2×10^{-6}**	**1×10^{-5}**

INL= Idaho National Laboratory; LCF = latent cancer fatality; MEI = maximally exposed individual; PuO_2 = plutonium oxide; SRS = Savannah River Site; VTR = Versatile Test Reactor.

[a] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[b] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[c] Risks are the product of the probability and impacts. For purposes of this EIS, events categorized as "unlikely" are assigned a probability of 10^{-2} Events categorized as "extremely unlikely" are assigned a probability of 10^{-4}. Events categorized as "extremely unlikely to beyond extremely unlikely" are assigned a probability of 10^{-5}. Events categorized as "beyond extremely unlikely" are assigned a probability of 10^{-6}.

[d] The total annual risk from reactor and support facility operations is the sum of all accidents evaluated. For the MEI, it represents a reasonable estimate of the annual risk that the MEI might receive sufficient dose to result in a LCF. For the population within 50 miles, it represents a reasonable estimate of the annual number of LCFs that might occur within that population due to accidents.

D.4.8 Comparison of the Annual Radiological Risks from Accidents at Versatile Test Reactor-Related Activities at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site to Other Radiological Risks

Table D–31 summarizes the total annual risk from reactor and support facility operations at INL, ORNL, and SRS, taking into account the probability of occurrence of each accident. For each option, the values are simply the sum of the LCF risk of all accidents evaluated. For the MEI, the summary represents a reasonable estimate of the annual risk that the MEI might receive sufficient dose to result in a LCF. For the population within 50 miles, the summary represents a reasonable estimate of the annual number of LCFs that might occur within that population due to accidents.

Table D–31. Summary of the Total Annual Risks for VTR-Related Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

	MEI (annual LCF risk) [a,b]	*Population (Near+Long-Term) (annual person-rem risk)* [a,b,c,d]	*Population (Near+Long-Term) (annual LCF risk)* [a,b,c,d]
Total Risk: Reactor and Support Operations at INL	3×10^{-9}	4×10^{-2}	3×10^{-5}
Total Risk: Reactor and Support Operations at ORNL	2×10^{-7}	2×10^{-1}	1×10^{-4}
Total Risk: Reactor Fuel Production Operations at INL	8×10^{-9}	3×10^{-3}	2×10^{-6}
Total Risk: Reactor Fuel Production Operations at SRS	3×10^{-9}	2×10^{-2}	1×10^{-5}
VTR, Support, and Reactor Fuel Production Operations at INL	1×10^{-9}	5×10^{-2}	3×10^{-5}

INL= Idaho National Laboratory; LCF = latest cancer fatality; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; SRS = Savannah River Site; VTR = Versatile Test Reactor.

[a] Annual LCF risks presented in this table are the product of the probability of the accident occurring and its impacts. For purposes of this EIS, the probability of an event categorized as "unlikely" is 10^{-2}; and "extremely unlikely" is 10^{-4}. "Extremely unlikely to beyond extremely unlikely" is 10^{-5}; and "beyond extremely unlikely" is 10^{-6}.

[b] The total annual risk from reactor and support operations and the total annual risk for operations from reactor fuel production is the sum of all accidents evaluated. For the MEI, it represents a reasonable estimate of the annual risk that the MEI might receive sufficient dose from an accident to result in a LCF. For the population within 50 miles, it represents a reasonable estimate of the annual number of LCFs that might occur within that population due to accidents.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

[d] Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage without mitigation measures, such as sheltering in place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passes. These doses include ingestion of contaminated foods, water, etc., direct exposure to deposited materials, and resuspension and inhalation of deposited materials. For purposes of the EIS, no interdiction or mitigation is assumed, but such measures would likely occur. The long-term risk reported includes both the near-term and long-term impacts without mitigation.

It is important to note that no near-term mitigation, such as emergency notifications to shelter in place or evacuate, or long-term restrictions on access to contaminated areas and interdiction of food supplies are assumed in evaluating the consequences.

Potential MEI Risks from VTR Activities at INL, ORNL, and SRS

The potential annual risks to the hypothetical MEI are low at all sites. For VTR and support operations at INL and ORNL, the MEI annual risks are 3×10^{-9} and 2×10^{-7} LCFs per year, respectively. The principal difference is the MEI for ORNL is assumed to be a boater on Melton Hill Lake within the overall ORO property boundary. The nearest residence is much farther away.

For VTR fuel production operations at INL and SRS, the potential annual MEI risks are 8×10^{-9} and 3×10^{-9} LCFs per year, respectively. In this case, the MEI for SRS is further away than at INL.

Potential Population Risks from VTR Activities at INL, ORNL, and SRS

The potential annual risks to the populations within 50 miles are low at all sites. For VTR and support operations at INL and ORNL, the annual population risks without the hypothetical beyond-design-basis reactor accident are 3×10^{-5} and 1×10^{-4} LCFs per year, respectively. The principal difference in the population risk is a higher population density near ORNL than near INL.

For VTR fuel production operations at INL and SRS, the potential annual population risks are 2×10^{-6} and 1×10^{-5} LCFs per year, respectively. The principal difference in the population risk is a higher population density near SRS than near INL.

If both VTR and reactor fuel production operations were at INL, the near-term plus long-term annual population risk would be 3×10^{-5} LCFs.

D.4.9 Versatile Test Reactor Beyond-Design-Basis Reactor Accidents

By design, the VTR is able to withstand a wide range of accidents. Most events that could affect safe operation of the VTR are mitigated by the VTR design. This section addresses potential beyond-design-basis accidents that have the potential for high consequences even though the probability is very low (1×10^{-6} to 1×10^{-8} per year). These accidents represent events in which the consequences can be in the hundreds or thousands of rem to the public while probabilities are less than one in a million per year. Consideration of these very low-probability but potentially high-consequence accidents provides valuable insight for the public and decision-makers in understanding the overall risks of operation, siting decisions, and the need for emergency preparedness.

Deterministic safety analyses for the VTR, based on the conceptual design (INL 2019, 2020c) and the preliminary VTR PRA (GEH 2019), are ongoing and are expected to continue to evolve with the VTR design. As part of the on-going safety analysis process (INL 2019), a more elaborate VTR PRA with a larger spectrum of events than currently quantified in terms of both probability and consequences is needed. These events will include the full spectrum of design-basis events and beyond-design-basis events, including accidents initiated by human failures, natural phenomena, and external events. Preliminary results from the ongoing PRA for the VTR conceptual design do not quantify the probabilities and consequences for beyond-design-basis accident sequences. Given that the time is not ripe to prepare a full VTR PRA on a yet-to-be finalized VTR design, hypothetical, beyond-design-basis accidents are postulated for the VTR in order to fulfill the requirements of NEPA.

D.4.9.1 Potential Hypothetical Beyond-Design-Basis Accident Scenarios

The purpose of considering these scenarios is to ensure that events that have potentially large offsite consequences are sufficiently understood and properly categorized. This process typically leads to design changes to prevent or mitigate any of these events. For the VTR EIS, a range of potential operational, external, and natural-phenomena hazards have been considered that might lead to severe accident conditions, including

- Human failures
- Plant design or construction errors
- Aircraft crash
- Seismic hazards
- Extreme straight-line wind, tornado, and hurricane hazards
- Flood, seiche, tsunami, and other flood-related hazards
- Extreme precipitation hazards
- Volcanic eruption hazards.

In typical power reactors, radiological risks to the public are dominated by loss-of-cooling, reactor-core-damage, and failure-of-confinement events with estimated frequencies in the one in ten thousand to one in a hundred thousand per year range. The safety analyses for the VTR conceptual design indicate that the pool-type, sodium-cooled design of the VTR and other design features provide substantial protection to the VTR from the severe accidents that would lead to core-melt and containment-failure accidents in commercial LWR power plants.

The hypothetical, beyond-design-basis reactor accident with loss of cooling is considered in this VTR EIS to provide a reasonable but bounding estimate of the potential impacts from very low-probability, high-consequence accidents. The probability of this event is estimated to be ~10^{-7} or less. As the VTR design evolves past the conceptual design phase, additional event initiators and subsequent accident sequences may be developed but this postulated, hypothetical event is expected to provide a reasonable estimate of the impacts of such an event. This event would bound a possible intentionally destructive act involving an aircraft.

D.4.9.2 Material at Risk for Severe Accidents in the Versatile Test Reactor

The MAR is based on fuel assemblies with 6 percent burnup. Current projections indicate that fuel burnup will not exceed 6 percent. The reactor vessel is designed to contain 110 fuel assemblies of which 66 fuel assemblies are in the reactor core and 44 used fuel assemblies are in out-of- core storage. The 66 in-core fuel assemblies are assumed to have no decay while the 44 out-of-core fuel assemblies are assumed to have decayed for 220 days.

By necessity, the assumptions used in modeling the impacts of this hypothetical event are quite conservative. Chemical and physical properties that would be expected to mitigate and limit a release are not considered. All of the active and passive cooling is assumed to be lost and all of fuel in the reactor is assumed to melt and be released quickly. Because the release is assumed to occur very quickly, no decay of even very short-lived isotopes is assumed. These very short-lived isotopes, with half lives of seconds to minutes or even hours contribute substantially to close-in doses. In reality, the major portion of a release, even with total loss of cooling, may not occur for hours or days after reactor shut down. In contrast, severe LWR accidents such as loss-of-cooling accidents typically assume that the major releases to the environment occur many hours or days after the initial event shutting off the reactor, allowing time for the short-lived isotopes to decay and for emergency actions, such as evacuation, to be implemented.

D.4.9.3 Release Fractions for Severe Accidents in the Versatile Test Reactor

For most VTR accidents in which fuel fails and fission products are released, nongaseous fission products would be retained in the sodium (Bucknor et al. 2017; Brunett et al. 2016; Grabaskas et al. 2016a). However, if bulk boiling of the sodium were to occur, retention of radionuclides in the sodium may be limited. The actual airborne release fractions and respirable fractions by isotope group for specific VTR events are unknown. Various ARFs and RFs have been assumed in the past. More recently, there have been efforts to better understand the potential phenomena during severe accidents and develop mechanistic source terms for severe accidents at sodium-cooled fast reactors. An ongoing program at Argonne National Laboratory is researching mechanistic source terms for fast reactors (Brunett et al. 2016; Clark et al. 2017; Grabaskas et al. 2015, 2016a, 2016b, 2017; Middleton et al. 2011) and sophisticated modeling codes have been developed.

The fraction of radionuclides in VTR fuel that might ultimately be released to the environment is speculative, but, if all active and passive cooling failed, could be substantial. Experimental data for releases of various isotopes from metal fuel in a pool of sodium has been reviewed and used to postulate mechanistic release fractions from a metal fuel, pool-type sodium fast reactor for various temperature ranges that might occur under operating and accident conditions (Grabaskas et al. 2016a). Potential release fractions and uncertainty estimates for key isotopic groups are postulated for normal operation temperatures (~500 °C), eutectic formation temperatures (~700 °C), fuel melting temperatures (~1100 °C), and very high temperatures (≥~1300 °C). **Table D–32** summarizes the postulated release fractions and uncertainty estimates.

Other releases fractions were considered, including

- PRISM: During the early licensing efforts for PRISM site suitability, the initial PRISM source term assumed complete core damage (including the spent fuel in the storage area) because of the lack of data on the behavior of metal fuel in severe liquid-metal-reactor accidents. The analysis did not include retention of radionuclides in the sodium pool. The near instantaneous release to confinement was modeled. The source term used for the site suitability assessment was 100 percent of the noble gas inventory, 0.1 percent of halogens, 0.1 percent of the volatiles, and 0.01 percent of the transuranic nuclides (plutonium) (PSID 1987).
- GNEP Draft EIS-0396 (DOE 2008), Table D.1.4-1—*Release Parameters for Reactor Beyond Design Basis Earthquakes and Aircraft Crashes* used noble gases (xenon, krypton) 1.0; halogens (iodine, bromine) 0.4; alkali metals (cesium, rubidium) 0.3; tellurium metals (tellurium, antimony, selenium) 0.05; barium, strontium 0.02; noble metals (ruthenium, rhodium, palladium, molybdenum, technetium, cobalt) 2.5×10^{-3}; lanthanides (lanthanum, zirconium, neodymium, niobium, promethium, samarium, yttrium, curium, americium) 2×10^{-4} based on Table 1 of NRC Regulatory Guide 1.183 (NRC 2000).
- Clinch River Breeder Reactor (CRBR): For purposes of determining the suitability of the proposed site for construction and operation of the CRBR in accordance with 10 CFR Part 100, a hypothetical core disruptive accident was evaluated. The radiological source term associated with this hypothetical release was specific in terms of percentages of fission products and fuel material released from the core to the reactor confinement building. The source term used for site suitability assessment was 100 percent of noble gas inventory. The inventory included 50 percent halogen (25 percent airborne), 1 percent solid fission product, and 1 percent plutonium inventory (PMC 1976). The PSAR further states that "The source term specified by NRC not only envelopes all design basis accidents considered in Chapter 15, but further envelopes a wide range of conservatively hypothesized core-related events. Evidence, both analytical and experimental, supports the applicant's position that compliance with the requirements of 10 CFR 100 could be demonstrated with a less stringent source term" (PMC 1976).

For purposes of this VTR EIS, the release fractions in Table D–32 for the fuel melting region of ~1,100 degrees Celsius are assumed for the fuel and fission products. These release fractions are higher than presented above for the 1987 PRISM analyses (PSID 1987) and are more experiment-based than those assumed for the CRBR analyses (PMC 1976). Except for the lanthanides, these release fractions for the most dose-significant isotope groups are similar to what might be expected for core-melt with containment failure accidents with LWRs. At both INL and ORNL, the lanthanides are a major contributor to the population doses. The lanthanide release fractions are about 30 percent for a sodium-cooled reactor with fuel melting accident and about 1 to 1.5 percent (Grabaskas et al. 2015) for a loss-of-coolant severe accident in an LWR. As a result, without allowance for decay, the lanthanides contribute about 74 and 63 percent of the near- and long-term population dose at ORNL, and about 77 and 22 percent of the near- and long-term population dose at INL. As a sensitivity analysis, source terms and impacts are also evaluated assuming the release fractions for VTR fuel are in the very-high-temperature region (≥~1,300 degrees Celsius). The resulting source terms would have the potential for impacts several times higher than projected for the fuel-melting region (~1,100 degrees Celsius).

Table D–32. Release Fractions for Isotope Groups for Metal-Fuel Sodium Fast Reactors

Isotope Group [a]	Release Fraction and Uncertainty Estimate							
	Normal Operation		Eutectic Formation		Fuel Melting		Very High Temperatures	
	Temperature ~500°C	Uncertainty	Temperature ~700°C	Uncertainty	Temperature ~1100°C	Uncertainty	Temperature ≥ 1300°C	Uncertainty
Noble Gases Group: (xenon, krypton)	≤ 0.85	Low	≤ 1	Medium	~ 1	Low	~ 1	Low
Halogens Group: (iodine, bromine)	≤ 0.15	Medium	≤ 0.20	Medium	≤ 0.3	Medium	≤ 1	Low
Alkali Metals Group: (cesium, rubidium)	≤ 0.55	Low	≤ 0.6	Medium	≤ 1	Medium	≤ 1	Low
Tellurium Group: (tellurium, antimony, selenium)	≤ 0.01	Medium	≤ 0.01	Medium	≤ 0.05	High	No Data	No Data
Barium, Strontium Group: (barium)	≤ 0.05	Medium	≤ 0.1	Medium	≤ 0.15	High	≤ 0.2	Medium
Barium, Strontium Group: (strontium)	≤ 0.001	Medium	≤ 0.05	Medium	≤ 0.2	High	≤ 0.2	High
Noble Metals Group: (ruthenium, rhodium, palladium, molybdenum, technetium, cobalt)	≤ 0.001	Low	≤ 0.01	Medium	≤ 0.05	Medium	≤ 0.05	Medium
Lanthanides Group [a]: (lanthanum, zirconium, neodymium, niobium, promethium, praseodymium, samarium, yttrium, curium, americium)	≤ 0.001	Medium	≤ 0.01	High	≤ 0.3	High	≤ 0.3	High
Lanthanides Group [a]: (europium)	≤ 0.55	Low	≤ 0.6	Medium	≤ 1	Medium	≤ 1	Low
Cerium Group [a]: (cerium)	≤ 0.01	Medium	≤ 0.05	High	≤ 0.1	High	≤ 0.15	High
Cerium Group [a]: (uranium, plutonium, neptunium)	≤ 0.001	Low	≤ 0.001	Medium	≤ 0.001	Medium	≤ 0.001	Medium

[a] Elements are grouped by general chemical characteristics typically used for development of reactor accident releases. For this analysis, the reference Grabaskas et al. 2016a further divided some of the isotope groups because of the specific melting characteristics of the elements at these temperatures.

Source: Adapted from Sections 5.1.1–4.1.8 of Grabaskas et al. 2016a.

D.4.9.4 Potential Releases from a Hypothetical, Beyond-Design-Basis Reactor Accident with Loss of Cooling

In order to fulfill the requirements of NEPA and for VTR EIS purposes, potential releases and impacts are evaluated for a hypothetical beyond-design-basis reactor accident of unknown cause in which all active (heat removal system [HRS]) and passive (RVACS) cooling systems are disrupted. For the sequence of events with total loss of heat removal capabilities, that is loss of both RVACS and HRS or the loss of heat sink, bulk sodium boiling and release of radionuclides from melted fuel in the reactor core is assumed. Because both the VTR reactor vessel and reactor room are not designed to withstand pressurization due to bulk sodium boiling, the confinement systems are assumed to fail (GEH 2019:117).

If a beyond-design-basis reactor accident with loss of cooling were to occur, the releases are assumed to occur initially at ground level but to rise as an elevated plume due to the residual heat of the reactor core and boiling sodium. For evaluation purposes, modeling was performed for a range of potential feasible heats and impacts associated with the highest population impacts are reported.

D.4.9.5 Potential Impacts and Risks from a Hypothetical Beyond-Design-Basis Reactor Accident at Idaho National Laboratory and Oak Ridge National Laboratory

Potential radiological impacts and risks to the MEI and public from the hypothetical beyond-design-basis accident are presented in **Tables D–33** and **D–34**. As expected, without allowing for pre-release decay and emergency actions, the results of the MACCS modeling indicate very high, likely fatal doses near the reactor site. An individual remaining at the assumed location of the MEI for the entire plume passage would receive a fatal dose. Individuals, including members of the public that remained near the reactor site, could receive very high and potentially fatal doses.

At INL, the projected public population within 10 miles of the VTR is less than 500, while at ORNL it is about 160,000. This makes a large difference in the projected unmitigated LCFs among the population, with MACCS projections of less than 10 for the INL site, but 3,200 for the ORNL site.

The combined effects of the conservative accident and modeling assumptions result is an over statement of the potential hypothetical VTR accident by a substantial amount. For example, evacuation at ORNL out to 10 miles could reduce the LCFs from ~8,400 (0-50 miles) by 3,200 (0-10 miles), leaving ~5,200 LCFs. The INL location is sufficiently remote that evacuation would not result in a substantial reduction in impacts.

The radiological impacts on the population are dominated by the early release of fission products from the reactor core and not the longer-term doses from ingestion of contaminated food. With the release fractions for fuel melting from Table D–32 assumed, the early doses are dominated by radioisotopes from the lanthanides (75 percent of the dose), halogens (8 percent of the dose), and barium/strontium (6 percent of the dose). About 30 percent of the lanthanides are assumed to be released from the molten fuel due to their relatively lower melting temperature as compared to the fuel itself.

Potential impacts on the offsite populations are proportionally higher for the reactor accident with loss of cooling than for other VTR accidents due to the increased MAR, the potentially higher airborne release fractions if active and passive heat removal were disrupted and sodium boiling were to occur, and the availability of short-lived isotopes within the reactor core. For the beyond-design-basis reactor accident with loss of cooling, the consequences could also be several orders of magnitude greater than those calculated for the highest-consequence design-basis earthquake for either the VTR or its support facilities.

Table D–34 presents the annual risks to the MEI and public from the unmitigated, hypothetical beyond-design-basis reactor accident. These risks are calculated by multiplying the projected impacts from Table D–33 by the estimated probability of the accident of 1×10^{-7} per year. Even without mitigation, these LCF risks are quite small.

Table D–33. Impacts for an Unmitigated, Hypothetical Beyond-Design-Basis Reactor Accident at Idaho National Laboratory and Oak Ridge National Laboratory

Location	*Frequency (per year)*	*Impacts on Noninvolved Worker (330 feet)*		*Impacts on an MEI – No Mitigation*		*Near-Term Impacts on Population within 50 Miles – No Mitigation*		*Near + Long-Term Impacts on Population within 50 Miles – No Mitigation*	
		Dose (rem) [a]	*Probability of an LCF* [b]	*Dose (rem)* [a]	*Probability of an LCF* [b]	*Early Dose (person rem)* [a]	*LCFs* [c]	*Early + Chronic Dose (person rem)* [a]	*LCFs* [c]
INL	Hypothetical – Estimated ~10^{-7} or less	5.2×10^{5} [d]	1 [d]	7.9×10^{2} [d]	1 [d]	1.3×10^{5} [d]	220 [d]	1.9×10^{5} [d]	260 [d]
ORNL		1.3×10^{6} [d]	1 [d]	4.8×10^{4} [d]	1 [d]	7.2×10^{6} [d]	8,000 [d]	7.9×10^{6} [d]	8,400 [d]

INL= Idaho National Laboratory; ORNL=Oak Ridge National Laboratory; LCF = latent cancer fatality; MEI = maximally exposed individual; rem = roentgen equivalent man; VTR = Versatile Test Reactor.

[a] Calculated using the source terms derived from Sections D.4.9.2 and D.4.9.3. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less. These results are the MACCS-calculated LCFs.

[d] For NEPA purposes, a hypothetical, beyond-design-basis reactor accident with loss of cooling is included in this VTR EIS to provide a reasonable but bounding estimate of the potential impacts from very low-probability, high-consequence accidents. The probability of this event is estimated to be ~10^{-7} or less. As the VTR design evolves past the conceptual design phase, additional event initiators and subsequent accident sequences may be developed. Accident probabilities and consequences are quantified as a part of the future VTR safety analysis and probabilistic risk assessment processes.

Table D–34. Annual Risks for an Unmitigated, Hypothetical Beyond-Design-Basis Reactor Accident at Idaho National Laboratory and Oak Ridge National Laboratory

Accident Scenario	*Frequency (per year)*	*MEI LCF Risk – No Mitigation (Probability of an LCF)* [a]		*Population LCF Risk – No Mitigation (Near-Term) (LCFs)* [a]		*Population LCF Risk – No Mitigation (Near+Long-Term) (LCFs)* [a]	
		INL	*ORNL*	*INL*	*ORNL*	*INL*	*ORNL*
VTR Accident LCF Risks							
Hypothetical Beyond-Design-Basis Reactor Accident with Loss of Cooling [b]	Hypothetical - Estimated ~10^{-7} or less	1×10^{-7}	1×10^{-7}	2×10^{-5}	8×10^{-4}	3×10^{-5}	8×10^{-4}

INL= Idaho National Laboratory; ORNL= Oak Ridge National Laboratory; LCF=latent cancer fatality; MEI = maximally exposed individual; rem = roentgen equivalent man; VTR = Versatile Test Reactor.

[a] Calculated using the projected impacts in Table D–33.

[b] For NEPA purposes, a hypothetical, beyond-design-basis reactor accident with loss of cooling is included in this VTR EIS to provide a reasonable but bounding estimate of the potential impacts from very low-probability, high-consequence accidents. The probability of this event is estimated to be ~10^{-7} or less. As the VTR design evolves past the conceptual design phase, additional event initiators and subsequent accident sequences may be developed. Accident probabilities and consequences are to be quantified as a part of the future VTR safety analysis and probabilistic risk assessment processes.

D.4.9.6 Comparison to U.S. Nuclear Regulatory Commission-Licensed Power Reactor Risks

Although the VTR would be regulated by the DOE and not the NRC,[6] comparing the potential VTR accident risks to those of LWRs regulated by the NRC provides insight into the relative risks from the VTR. The NRC has recently released an EIS (NRC 2019a; NUREG-2226) that includes a summary of environmental risks from severe accidents for current nuclear power plants. Table 5-18 of that EIS summarizes the core damage frequencies and 50-mile population dose risk. The table indicates that, based on over 70 current plants at over 40 sites, the current mean reactor core damage frequency is 3.1×10^{-5} per year with an annual 50-mile population dose risk of 15 person-rem (or ~2×10^{-2} LCFs) per reactor. The range of 50-mile population dose risk for current operating power reactors is 0.55 to 69 person-rem per year, or approximately 7×10^{-4} to 8×10^{-2} LCFs per year. All of the NRC estimates were modeled with early evacuation and calculated using MACCS2.

Table D–35 compares the annual population risks from operation of the VTR to that of commercial LWRs. It is important to note that the VTR risks are based on conservative assumptions that do not consider decay of short-lived isotopes, mitigation to limit releases, or emergency actions such as evacuation or sheltering-in-place. Thus the potential VTR impacts are likely over stated. On the other hand, the NRC-evaluated risks are based on more realistic assumptions that consider preventative and mitigation features of the LWRs, including evacuation of persons within the typical 10-mile radius emergency planning zones surrounding the LWRs. Severe accident modeling for LWRs also considers for radioisotope decay for releases that occur hours or days after the reactor shuts down.

Table D–35. Summary of the Total Annual Risks for LWR and VTR Severe Accidents

Severe Accidents with VTR and LWRs	*Severe Accident Frequency (per year)* [a,b]	*50-Mile Population Risk (person-rem per year)* [a,b]	*50-Mile Population Risk (LCF risk per year)* [a,b,c]
Unmitigated, Hypothetical Beyond-Design-Basis Reactor Accident: INL	Hypothetical - Estimated ~10^{-7} or less	0.019	3×10^{-5}
Unmitigated, Hypothetical Beyond-Design-Basis Reactor Accident: ORNL	Hypothetical - Estimated ~10^{-7} or less	0.79	8×10^{-4}
Highest Risk: Upper Range of NRC LWRs – with Evacuation	2.4×10^{-4}	69	8×10^{-2}
Mean Risk: NRC LWRs – with Evacuation	3.1×10^{-5}	15	2×10^{-2}
Lowest Risk: Lower Range of NRC LWRs – with Evacuation	1.9×10^{-6}	0.55	7×10^{-4}

INL= Idaho National Laboratory; ORNL= Oak Ridge National Laboratory; LCF=latent cancer fatality; MEI = maximally exposed individual; rem = roentgen equivalent man; VTR = Versatile Test Reactor, NRC=Nuclear Regulatory Commission.

[a] VTR risk calculated using the projected impacts in Table D–33.

[b] LWR data from Table 5018 of NUREG-2226 (NRC 2019a).

[c] LCF risks for VTR calculated from projected LCFs in MACCS. LCF risks for LWRs calculated by multiplying reported person-rem per year in NRC 2019a, Table 5-18 by 2 times 0.0006 LCFs per person-rem.

[6] Under the Atomic Energy Act (AEA) of 1954 and its amendments, and the AEA Energy Reauthorization Act (ERA) of 1974, DOE has the authority to develop, construct and operate its own reactors. Under this authority, DOE plans to conduct the safety review for the VTR and authorize its construction and operation. DOE research reactor facilities, such as the VTR, are exempt from Nuclear Regulatory Commission (NRC) licensing in accordance with Section 110 of the ERA and Title 10, Code of Federal Regulations 50.11, Exceptions and Exemptions from Licensing Requirements. The VTR Project will consider guidance developed for non-LWRs, such as NRC Regulatory Guide 1.232, Guidance for Developing Principal Design Criteria for non-Light Water Reactors, and the recently developed NEI 18-04, Risk-Informed Performance-Based Technology Guidance for non-Light Water Reactors, in the development of VTR safety basis documentation.

Table D–35 indicates that the 50-mile person-rem risk for the VTR is much smaller than that of the typical LWR. A portion of this reduction is due to lower power levels in the VTR versus a LWR, but the major factor leading to the reduction is the much lower annual core damage frequency, on the order of 10^{-7} or less for the VTR versus 3.1×10^{-5} for the mean risk LWR.

D.4.9.7 Comparison of Versatile Test Reactor Hypothetical Beyond-Design-Basis Accident Risk to DOE Nuclear Safety Policy

To provide further insight into the safety of the VTR, the VTR risk estimates for the unmitigated, hypothetical, beyond-design-basis accident are compared to the DOE nuclear safety policy and Safety Goals for all DOE Operations (DOE 2011, DOE P 420.1).

DOE Nuclear Safety Policy and Safety Goals

DOE has established a nuclear safety policy (DOE P 420.1) and established safety goals for the conduct of its operations. The DOE Safety Goal is "to conduct its operations such that (a) Individual members of the public be provided a level of protection from the consequences of DOE operations such that individuals bear no significant additional risk to life and health to which members of the general population are normally exposed, and (b) DOE workers' health and safety are protected to levels consistence with or better than that achieved for workers in similar industries."

The following two quantitative safety objectives for public protection are established as "aiming points" (not requirements) in support of the Safety Goal that guides the development of DOE's nuclear safety requirements and standards:

- The risk to an average individual in the vicinity of a DOE nuclear facility for prompt fatalities that might result from accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the population are generally exposed. For evaluation purposes, individuals are assumed to be located within one mile of the site boundary (DOE P 420.1).
- The risk to the population in the area of a DOE nuclear facility for cancer fatalities that might result from operations should not exceed one-tenth of one percent (0.1%) of the sum of all cancer fatality risks resulting from all other causes. For evaluation purposes, individuals are assumed to be located within 10 miles of the site boundary (DOE P 420.1).

NRC Safety Goals

Even though the VTR is not subject to NRC regulation, consideration of the NRC safety goals provides additional assurance that the VTR can be safely operated. The NRC has set safety goals for average individual early fatality and LCF risks from reactor accidents in the Safety Goal Policy Statement (51 FR 30028). The Safety Goal Policy Statement expressed the Commission's policy regarding the acceptance level of radiological risk from nuclear power plant operation as follows (NRC 2019a):

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative health objectives are used in determining achievement of the NRC safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

The two DOE quantitative safety objectives for public are similar to the NRC quantitative health objectives. NRC interpreted the two quantitative health objectives by translating them into two numerical objectives (NRC 2019a):

- The individual risk of a prompt fatality from all "other accidents to which members of the U.S. population are generally exposed," is about 4.0×10^{-4} per year, including a 1.3×10^{-4} per year risk associated with transportation accidents; one-tenth of 1 percent of these figures implies that the individual risk of prompt fatality from a reactor accident should be less than 4×10^{-7} per reactor-year (NRC 2019a).
- "The sum of cancer fatality risks resulting from all other causes" for an individual is taken to be the cancer fatality rate in the United States, which is about 1 in 500 or 2×10^{-3} per year; one-tenth of 1 percent of this implies that the risk of cancer to the population in the area near a nuclear power plant because of its operation should be limited to 2×10^{-6} per reactor-year (NRC 2019a).

Since the DOE and NRC safety objectives are the same, the NRC quantitative risk numbers can be used for both DOE and NRC comparisons.

Comparison VTR Risks to DOE and NRC Safety Goals

As indicated above, the quantitative individual risk objective or goal and the population objective or goal for both DOE nuclear facility operations and NRC power plant operations are essentially identical and can be used for comparison to VTR risks. MACCS2 calculates average individual early fatality and LCF risks. The average individual early fatality risk is calculated using the population distribution within 1 mile of the site boundary. The average individual LCF risk is calculated using the population distribution within 10 miles of the site.

<u>Individual Risk Goal Comparisons</u>**:** For the VTR at INL, the average individual (within I mile of the site boundary) risk of a prompt facility from the hypothetical beyond-design-basis reactor accident is 2×10^{-2} early fatalities × 1×10^{-7} accidents per year or 2×10^{-9} early fatalities per reactor year. At ORNL, the early fatality risk is 8×10^{-2} early fatalities × 1×10^{-7} accidents per year or 6×10^{-9} early fatalities per reactor year. Thus the early fatality risk from the hypothetical beyond-design-basis accident is 0.60 percent and 1.5 percent, respectively, of the DOE and NRC safety goals for prompt fatalities.

<u>Population Risk Goal Comparisons</u>**:** For the VTR at INL, the average individual (within 10 miles of the site boundary) risk of an LCF from the hypothetical beyond-design-basis reactor accident is 1×10^{-2} LCFs × 1×10^{-7} accidents per year or 1×10^{-9} LCFs per reactor year. At ORNL, the LCF risk is 2×10^{-2} LCFs × 1×10^{-7} accidents per year or 2×10^{-9} LCFs per reactor year. Thus the LCF risk from the hypothetical beyond-design-basis accident is 0.06 percent and 0.10 percent, respectively, of the DOE and NRC safety goals for LCFs.

DOE Safety Goal Conclusion: The VTR sited at either INL or ORNL would meet the DOE the safety goals. The safety goals for prompt fatalities or latent cancers would be met by a wide margin, even with the many conservative assumptions used in the accident source term and impact calculations. It is expected that as the VTR design progresses and a seismic PRA is conducted, the design will evolve to demonstrate that the hypothetical accident postulated here both overstates the potential damage from beyond extremely unlikely events, such as a far beyond-design-basis earthquake and the likelihood of such an event.

D.4.9.8 Economic Costs of Severe Accidents

The most severe accident postulated, in terms of potential radiological consequences, is the unmitigated, hypothetical beyond-design-basis reactor accident with loss of cooling. It is expected that the ultimate design of the VTR would prevent or substantially mitigate releases if an event occurred that had the potential for partial or total loss of cooling of the reactor. Thus, both the estimated probability of this event, one in ten million or less per year and the extent of damage to the reactor core and fraction of the core released to the environment should be quite conservative. If the event were initiated by a severe earthquake, the expected damage at MFC, INL, and the region would be extensive with most structures (homes, businesses, and infrastructure) severely damaged. Initial emergency response to the earthquake damage would likely focus on saving of lives.

The economic impacts of the hypothetical beyond-design-basis reactor accident with loss of cooling are speculative. The MACCS2 computer program, which is used for the accident impact evaluations, has the capability to project economic costs, including population-dependent costs, farm dependent costs, decontamination costs, interdiction costs, emergency phase costs, and milk and crop disposal costs. These economic models were developed by Sandia National Laboratory, the MACCS2 model developer, and the NRC. The models have been used for U.S. nuclear power plant evaluations for decades. Evaluations using this MACCS2 model incorporated INL and ORNL-specific regional data developed with the SECPOP companion computer code to MACCS2. The models projected economic costs within 50 miles for the severe accidents to be 290 and 3,500 million dollars at INL and ORNL, respectively. The models' projected economic costs for the ORNL region are much higher primarily due to the higher population density and the more varied land use in that area.

D.5 Hazardous Material Releases

D.5.1 Source Terms for Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

The hazardous material source terms for accidents at either SRS, INL, or ORNL are determined for a representative set of accidents. Accidents are selected from fuel fabrication, VTR operations, spent fuel handling and treatment, post-irradiation examination, and spent fuel cask operations. Plume rise is not considered for any of the fire scenarios. Using a ground-level release for the fires provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS.

D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure

The fire selected for analysis would affect the solid fuel material and sodium in the fuel fabrication line. The frequency of the earthquake is assumed to be extremely unlikely to beyond extremely unlikely. MAR is one assembly. All material in the fuel fabrication line would be at risk and the DR is 1. For sodium, the bounding ARF of 1×10^{-2} and an RF of 1 for combustible liquid is used (DOE 1994). The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively, are based on the airborne release of plutonium particulates formed by oxidation at elevated temperatures without self-sustained oxidation (DOE 1994). A building LPF of 1 is assumed. However, because the building debris would provide some confinement, a more realistic value is expected to be much lower. **Table D–36** summarizes MAR, release fractions, and source term for this accident scenario.

D.5.1.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure

A spill of melted spent fuel from the spent fuel treatment hot cell is assumed to occur when an earthquake causes failure of the hot cell's confinement. The frequency of such an earthquake is extremely unlikely. The hot cell enclosure and the offgas exhaust ventilation system would be expected to fail and allow the release of hazardous material released in the spill. MAR is one assembly and the DR is 1. The bounding ARF and RF values of 2.0×10^{-5} and 1, respectively, are based on a free-fall spill (less than a 3-meter drop) of aqueous solutions with a density more than 1.2 gram per cubic centimeter (DOE 1994). There is no reduction in the source term because of the hot cell confinement, and an LPF of 1 is used. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

D.5.1.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure

The fire selected for analysis would affect the sodium and fuel material during spent fuel handling and treatment. The frequency of the sodium fire is extremely unlikely. MAR is one assembly, and the DR is 1. For sodium, the bounding ARF of 1×10^{-2} and an RF of 1 for combustible liquid is used (DOE 1994). The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively, are based on the airborne release of particulates formed by oxidation at elevated temperatures (DOE 1994). There is no reduction in the source term because of confinement, and an LPF of 1 is used. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

D.5.1.4 Fire Involving Test Assembly during Seismically-Induced Confinement Failure

The fire selected for analysis would impact the test assembly during post-irradiation examination. The frequency of the earthquake is extremely unlikely. MAR is one pin of a fuel assembly, and the DR is 1. For sodium, the bounding ARF of 1×10^{-2} and an RF of 1 for combustible liquid are used (DOE 1994). The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively are based on the airborne release of particulates formed by oxidation at elevated temperatures (DOE 1994). There is no reduction in the source term because of the fuel fabrication line's confinement, and an LPF of 1 is used. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

D.5.1.5 Drop of Fuel-Loaded Cask

After spent fuel has been washed, it is loaded into spent fuel storage casks pending transfer to the fuel treatment site. The accident is assumed to occur when a spent fuel storage cask is dropped during handling. Equipment failure or human error is assumed to cause the cask drop. The release is assumed to occur because of damage to the contents of one spent fuel storage cask. The release is assumed to occur at ground level over 10 minutes. The assumed accident frequency is extremely unlikely. MAR is the equivalent of six assemblies. The DR is 0.5 because not all fuel in the cask is expected to be involved. The ARF of 4×10^{-5} and an RF of 1 for aerodynamic entrainment and resuspension of surface contaminated waste are used (DOE 1994). The ARF and RF are 1 for sodium. The dropped cask is assumed to continue providing some confinement after the drop. Consequently, the LPF is assumed 0.5. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

Table D–36. Hazardous Material Source Terms

Accident	*Material at Risk*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Sodium Source Term (grams)*	*Uranium Source Term (grams)*
D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	1 Assembly	1	3×10^{-5}	0.04	1		0.042
		1	1×10^{-2}	1	1	9.5	
D.5.1.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	1 Assembly	1	2×10^{-5}	1	1	0.019	0.71
D.5.1.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	1 Assembly	1	3×10^{-5}	0.04	1		0.042
		1	1×10^{-2}	1	1	9.5	
D.5.1.4 Fire Involving Test Assembly (Seismically-Induced Confinement Failure)	1 Pin	1	3×10^{-5}	0.04	1		2.0×10^{-4}
		1	1×10^{-2}	1	1	0.044	
D.5.1.5 Drop of Fuel-Loaded Cask	6 Assemblies	0.5	4×10^{-5}	1	0.5	0.057	2.1
D.5.1.6 Sodium Fire due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building	4,000 kg Na	1	1×10^{-2}	1	1	40,000	-

Kg = kilogram; Na = sodium.

Notes: From Appendix B: 945.252 grams of sodium per assembly; 35,371 grams of uranium per assembly.

D.5.1.6 Sodium Fire due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building

The accident is assumed to occur when an earthquake causes a breach of the secondary heat removal system main branch piping. The flowrate in each loop is approximately 800 kilograms per second. A full breach of the piping in one loop is assumed to continue for five seconds. Thus, approximately 4,000 kilograms of sodium could be released and oxidized. The release is assumed to occur at ground level over 10 minutes. The assumed accident frequency is extremely unlikely. MAR is 4,000 kilograms of sodium. The DR is 1 because all of the sodium is assumed involved. The bounding ARF of 1×10^{-2} and an RF of 1 for combustible liquid is used (DOE 1994). Because the sodium is released outside of the reactor building, an LPF of 1 is assumed. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

D.5.2 Hazardous Material Impacts of Facility Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

Sodium and uranium are potential materials of concern for VTR-related activities. Hazardous material releases are evaluated by calculating the concentration of sodium hydroxide (the product formed by reaction of sodium with water) and uranium. The concentrations are compared to the DOE protective action criteria (PAC) to evaluate the hazard to human health. The concentration is calculated using the χ/q value from the site-specific MACCS2 calculation in conjunction with the release rate for the hazardous material. The release rate is calculated assuming that the quantity of the hazardous material is released over ten minutes (600 seconds). The resulting concentration is:

Concentration = Release Rate x χ/q

Where

Concentration	= hazardous material concentration (mg/m3 [milligrams per cubic meter])
Release Rate	= ST x (1000 milligrams per gram) per second)
ST	= source term (grams)
t	= release duration (seconds)
χ/q	= dispersion coefficient (second per cubic meter)

The ground-level dispersion coefficients from MACCS2 for INL, ORNL, and SRS are shown in **Table D–37**. The noninvolved worker is assumed to be downwind at a point of 330 feet from the accident. The MEI is located downwind at a point of 3.1 miles, 0.5 miles, and 5.5 miles from the accident at INL, ORNL, and SRS, respectively.

Table D–37. Ground Level Dispersion Coefficients χ/q

Site	Ground-Level Dispersion Coefficients χ/q (s/m^3)	
	Noninvolved Worker	Maximally Exposed Individual
INL	1.45×10^{-3}	1.84×10^{-6}
ORNL	3.81×10^{-3}	1.46×10^{-4}
SRS	3.41×10^{-3}	1.81×10^{-6}

INL= Idaho National Laboratory; ORNL = Oak Ridge National Laboratory; s/m^3 = seconds per cubic meter; SRS = Savannah River Site.

The PACs for sodium hydroxide and uranium are based on Emergency Response Planning Guidelines (ERPGs) and Temporary Emergency Exposure Limits (TEELs), respectively. The ERPGs and TEELs estimate the concentrations at which most people will begin to experience health effects if they are exposed to a hazardous airborne chemical for one hour. (Sensitive members of the public—such as children, seniors, or the chronically ill—are not covered by these guidelines. They may experience adverse effects at concentrations below the ERPG or TEEL values.) A chemical may have up to three ERPG or TEEL values, each of which corresponds to a specific tier of health effects. TEELs are intended for use until Acute Exposure Guideline Levels (AEGLs) or ERPGs are adopted for chemicals. The three ERPG tiers and TEEL tiers are defined as follows:

- ERPG-3 is the maximum concentration in air below which nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.
- ERPG-2 is the maximum concentration in air below which nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.
- ERPG-1 is the maximum concentration in air below which nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.
- TEEL-3 is the airborne concentration (expressed as ppm [parts per million] or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, when exposed for more than one hour, could experience life-threatening adverse health effects or death.
- TEEL-2 is the airborne concentration (expressed as ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, when exposed for more than one hour, could experience irreversible or other serious, long-lasting, adverse health effects or an impaired ability to escape.
- TEEL-1 is the airborne concentration (expressed as ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, when exposed for more than one hour, could experience notable discomfort, irritation, or certain asymptomatic, non-sensory effects. However, these effects are not disabling, but they are transient and are reversible upon cessation of exposure.

The ERPG values for sodium hydroxide and the TEEL values for uranium are shown in **Table D–38** (DOE 2018).

Table D–38. Emergency Response Planning Guideline Values for Sodium and Uranium

Sodium Hydroxide				***Uranium***		
ERPG-1 (mg/m³)	***ERPG-2 (mg/m³)***	***ERPG-3 (mg/m³)***		***TEEL-1 (mg/m³)***	***TEEL-2 (mg/m³)***	***TEEL-3 (mg/m³)***
0.5	5	50		0.6	5	30

ERPG = Emergency Response Planning Guidelines; mg/m^3 = milligrams per cubic meter; TEEL = Temporary Emergency Exposure Limit.

Sodium hydroxide (1.739 grams sodium hydroxide per gram of sodium) and uranium concentrations are presented in **Table D–39, D–40,** and **D-41** for INL, ORNL, and SRS (respectively) along with ERPG values, fractions of ERPG values, TEEL values, and fractions of TEEL values. Inspection of the concentrations in the tables shows that all of the uranium concentrations for the noninvolved worker are less than the TEEL-2 value for uranium. All of the uranium concentrations for the MEI are less than the TEEL-1 value for uranium. For the noninvolved worker and the MEI, the sodium hydroxide concentrations are less than the corresponding ERPG-2 and ERPG-1 values for sodium hydroxide for all events except the "Sodium Fire Due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building" event.

Table D–39. Sodium and Uranium Concentration for Accidents at Idaho National Laboratory

Accident	Sodium Release Rate (mg/s)	Uranium Release Rate (mg/s)	Noninvolved Worker: NaOH Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of ERPG-2 [a]	Noninvolved Worker: U Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of TEEL-2	Maximally Exposed Individual: NaOH Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of ERPG-1 [a]	Maximally Exposed Individual: Uranium Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of TEEL-1
D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	16	0.070	0.040 / 5 / 0.0081	1.0×10^{-4} / 5 / 2.0×10^{-5}	5.1×10^{-5} / 0.5 / 1.0×10^{-4}	1.3×10^{-7} / 0.6 / 2.1×10^{-7}
D.5.1.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	0.032	1.2	8.1×10^{-5} / 5 / 1.6×10^{-5}	0.0017 / 5 / 3.5×10^{-4}	1.0×10^{-7} / 0.5 / 2.0×10^{-7}	2.2×10^{-6} / 0.6 / 3.7×10^{-6}
D.5.1.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	16	0.070	0.040 / 5 / 0.0081	1.0×10^{-4} / 5 / 2.0×10^{-5}	5.1×10^{-5} / 0.5 / 1.0×10^{-4}	1.3×10^{-7} / 0.6 / 2.1×10^{-7}
D.5.1.4 Fire Involving Test Assembly During Seismically-Induced Confinement Failure	0.073	3.3×10^{-4}	1.8×10^{-4} / 5 / 3.7×10^{-5}	4.8×10^{-7} / 5 / 9.6×10^{-8}	2.3×10^{-7} / 0.5 / 4.7×10^{-7}	6.1×10^{-10} / 0.6 / 1.0×10^{-9}
D.5.1.5 Drop of Fuel-Loaded Cask	0.095	3.5	2.4×10^{-4} / 5 / 4.8×10^{-5}	0.0051 / 5 / 0.0010	3.0×10^{-7} / 0.5 / 6.1×10^{-7}	6.4×10^{-6} / 0.6 / 1.1×10^{-5}
D.5.1.6 Sodium Fire Due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building	67,000	-	170 / 5 / 34	-	0.21 / 0.5 / 0.43	-

ERPG = Emergency Response Planning Guidelines; HRS = heat removal system; mg/m^3 = milligrams per cubic meter; mg/s = milligrams per second; NaOH = sodium hydroxide; RVACS = Reactor Vessel Auxiliary Cooling System; TEEL = Temporary Emergency Exposure Limit.

[a] 1.739 grams NaOH per gram Na.

Table D–40. Sodium and Uranium Concentration for Accidents at Oak Ridge National Laboratory

Accident	Sodium Release Rate (mg/s)	Uranium Release Rate (mg/s)	Noninvolved Worker: NaOH Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of ERPG-2 [a]	Noninvolved Worker: Uranium Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of TEEL-2	MEI: NaOH Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of ERPG-1 [a]	MEI: Uranium Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of TEEL-1
D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	16	0.070	0.11 / 5 / 0.021	2.7×10^{-4} / 5 / 5.3×10^{-5}	0.0041 / 0.5 / 0.0081	1.0×10^{-5} / 0.6 / 1.7×10^{-5}
D.5.1.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	0.032	1.2	2.1×10^{-4} / 5 / 4.2×10^{-5}	0.0046 / 5 / 9.1×10^{-4}	8.1×10^{-6} / 0.5 / 1.6×10^{-5}	1.8×10^{-4} / 0.6 / 2.9×10^{-4}
D.5.1.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	16	0.070	0.11 / 5 / 0.021	2.7×10^{-4} / 5 / 5.3×10^{-5}	0.0041 / 0.5 / 0.0081	1.0×10^{-5} / 0.6 / 1.7×10^{-5}
D.5.1.4 Fire Involving Test Assembly During Seismically-Induced Confinement Failure	0.073	3.3×10^{-4}	4.8×10^{-4} / 5 / 9.7×10^{-5}	1.3×10^{-6} / 5 / 2.5×10^{-7}	1.9×10^{-5} / 0.5 / 3.7×10^{-5}	4.8×10^{-8} / 0.6 / 8.0×10^{-8}
D.5.1.5 Drop of Fuel-Loaded Cask	0.095	3.5	6.3×10^{-4} / 5 / 1.3×10^{-4}	0.013 / 5 / 0.0027	2.4×10^{-5} / 0.5 / 4.8×10^{-5}	5.1×10^{-4} / 0.6 / 8.5×10^{-4}
D.5.1.6 Sodium Fire Due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building	67,000	-	440 / 5 / 89	-	17 / 0.5 / 34	-

ERPG = Emergency Response Planning Guidelines; HRS = heat removal system; MEI = maximally exposed individual; mg/m^3 = milligrams per cubic meter; mg/s = milligrams per second; NaOH = sodium hydroxide; RVACS = Reactor Vessel Auxiliary Cooling System; TEEL = Temporary Emergency Exposure Limit.

[a] 1.739 grams NaOH per gram Na.

Table D–41. Sodium and Uranium Concentration for Accidents at Savannah River Site

Accident	Sodium Release Rate (mg/s)	Uranium Release Rate (mg/s)	Noninvolved Worker: NaOH Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of ERPG-2 [a]	Noninvolved Worker: Uranium Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of TEEL-2	MEI: NaOH Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of ERPG-1 [a]	MEI: Uranium Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of TEEL-1
D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	16	0.070	0.095 / 5 / 0.019	2.4×10^{-4} / 5 / 4.8×10^{-5}	5.0×10^{-5} / 0.5 / 1.0×10^{-4}	1.3×10^{-7} / 0.6 / 2.1×10^{-7}

ERPG = Emergency Response Planning Guidelines; MEI = maximally exposed individual; mg/m^3 = milligrams per cubic meter; mg/s = milligrams per second; NaOH = sodium hydroxide; SRS = Savannah River Site; TEEL = Temporary Emergency Exposure Limit.

[a] 1.739 grams NaOH per gram Na.

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D.7 Attachment D1: Isotopic Composition in Reactor-VTR Fuel and 220-Day Cooled VTR Fuel

Table D–42. Isotopic Composition In-Reactor and 220-Day Cooled VTR Fuel

Isotope	*Single 6% Burnup Assembly mass (grams)*	*Single 6% Burnup Assembly (curies)*	*VTR Assembly Decayed 220 Days mass (grams)*	*VTR Assembly Decayed 220 Days (curies)*
Am-241	$1.43{\times}10^{1}$	$4.91{\times}10^{1}$	$2.34{\times}10^{1}$	$8.04{\times}10^{1}$
Am-242	$1.95{\times}10^{-1}$	$1.58{\times}10^{5}$	0.00	0.00
Am-242m	0.00	0.00	0.00	0.00
Am-243	5.50	1.10	0.00	0.00
Am-244	$1.61{\times}10^{-4}$	$2.03{\times}10^{2}$	0.00	0.00
Am-245	$3.36{\times}10^{-9}$	$2.10{\times}10^{-2}$	0.00	0.00
Am-246	$7.53{\times}10^{-14}$	$1.48{\times}10^{-6}$	0.00	0.00
Ba-139	$2.04{\times}10^{-2}$	$3.34{\times}10^{5}$	0.00	0.00
Ba-140	3.44	$2.52{\times}10^{5}$	$2.20{\times}10^{-5}$	1.61
Ba-141	$4.23{\times}10^{-3}$	$3.09{\times}10^{5}$	0.00	0.00
Ba-142	$2.16{\times}10^{-3}$	$2.68{\times}10^{5}$	0.00	0.00
Br-83	$1.41{\times}10^{-3}$	$2.23{\times}10^{4}$	0.00	0.00
Br-84	$5.23{\times}10^{-4}$	$3.68{\times}10^{4}$	0.00	0.00
Ce-141	8.57	$2.44{\times}10^{5}$	$7.92{\times}10^{-2}$	$2.26{\times}10^{3}$
Ce-143	$3.87{\times}10^{-1}$	$2.57{\times}10^{5}$	0.00	0.00
Ce-144	$3.76{\times}10^{1}$	$1.20{\times}10^{5}$	$2.20{\times}10^{1}$	$7.02{\times}10^{4}$
Cm-242	$6.07{\times}10^{-1}$	$2.01{\times}10^{3}$	$3.02{\times}10^{-1}$	$1.00{\times}10^{3}$
Cm-243	$5.34{\times}10^{-3}$	$2.76{\times}10^{-1}$	$5.26{\times}10^{-3}$	$2.72{\times}10^{-1}$
Cm-244	$4.16{\times}10^{-1}$	$3.37{\times}10^{1}$	$4.08{\times}10^{-1}$	$3.30{\times}10^{1}$
Cm-245	$9.79{\times}10^{-3}$	$1.68{\times}10^{-3}$	$9.79{\times}10^{-3}$	$1.68{\times}10^{-3}$
Cm-246	$1.49{\times}10^{-4}$	$4.58{\times}10^{-5}$	$1.49{\times}10^{-4}$	$4.58{\times}10^{-5}$
Cs-134	1.70	$2.20{\times}10^{3}$	1.39	$1.80{\times}10^{3}$
Cs-135	$9.98{\times}10^{1}$	$1.15{\times}10^{-1}$	$1.00{\times}10^{2}$	$1.15{\times}10^{-1}$
Cs-136	$1.08{\times}10^{-1}$	$7.92{\times}10^{3}$	$1.00{\times}10^{-6}$	$7.33{\times}10^{-2}$
Cs-137	$8.83{\times}10^{1}$	$7.69{\times}10^{3}$	$8.71{\times}10^{1}$	$7.58{\times}10^{3}$
Eu-152m	0.00	0.00	0.00	0.00
Eu-154	$7.33{\times}10^{-1}$	$1.93{\times}10^{2}$	$6.98{\times}10^{-1}$	$1.84{\times}10^{2}$
Eu-155	2.52	$1.17{\times}10^{3}$	2.31	$1.07{\times}10^{3}$
Eu-156	$1.41{\times}10^{-1}$	$8.08{\times}10^{4}$	$6.30{\times}10^{-6}$	3.61
I-129	$1.82{\times}10^{1}$	$3.21{\times}10^{-3}$	$1.82{\times}10^{1}$	$3.21{\times}10^{-3}$
I-130	$2.21{\times}10^{-3}$	$4.31{\times}10^{3}$	0.00	0.00
I-131	1.54	$1.91{\times}10^{5}$	$8.65{\times}10^{-9}$	$1.07{\times}10^{-3}$
I-132	$2.80{\times}10^{-2}$	$2.89{\times}10^{5}$	0.00	0.00
I-133	$3.31{\times}10^{-1}$	$3.75{\times}10^{5}$	0.00	0.00
I-134	$1.51{\times}10^{-2}$	$4.03{\times}10^{5}$	0.00	0.00
I-135	$1.01{\times}10^{-1}$	$3.55{\times}10^{5}$	0.00	0.00
Kr-83m	0.00	0.00	0.00	0.00
Kr-85	1.30	$5.10{\times}10^{2}$	$1.07{\times}10^{-5}$	$4.20{\times}10^{-3}$
Kr-85m	0.00	0.00	0.00	0.00
Kr-87	$2.43{\times}10^{-3}$	$6.88{\times}10^{4}$	0.00	0.00
Kr-88	$7.70{\times}10^{-3}$	$9.65{\times}10^{4}$	0.00	0.00
La-140	$4.40{\times}10^{-1}$	$2.45{\times}10^{5}$	$3.33{\times}10^{-6}$	1.85
La-141	$5.47{\times}10^{-2}$	$3.09{\times}10^{5}$	0.00	0.00
La-142	$1.91{\times}10^{-2}$	$2.73{\times}10^{5}$	0.00	0.00
La-143	$2.76{\times}10^{-3}$	$2.55{\times}10^{5}$	0.00	0.00
Mo-93	0.00	0.00	0.00	0.00

Isotope	***Single 6% Burnup Assembly mass (grams)***	***Single 6% Burnup Assembly (curies)***	***VTR Assembly Decayed 220 Days mass (grams)***	***VTR Assembly Decayed 220 Days (curies)***
Mo-99	6.98×10^{-1}	3.35×10^{5}	0.00	0.00
Nb-93m	0.00	0.00	0.00	0.00
Nb-94	1.61×10^{-4}	3.02×10^{-5}	1.61×10^{-4}	3.02×10^{-5}
Nb-95	5.96	2.33×10^{5}	1.10	4.30×10^{4}
Nb-96	1.36×10^{-3}	1.90×10^{3}	0.00	0.00
Nb-97	1.09×10^{-2}	2.93×10^{5}	0.00	0.00
Nb-97m	0.00	0.00	0.00	0.00
Nd-147	1.28	1.04×10^{5}	1.19×10^{-6}	9.63×10^{-2}
Nd-149	6.25×10^{-3}	7.60×10^{4}	0.00	0.00
Nd-151	4.51×10^{-4}	4.51×10^{4}	0.00	0.00
Np-237	1.09×10^{1}	7.69×10^{-3}	1.13×10^{1}	7.97×10^{-3}
Np-238	1.68×10^{-2}	4.35×10^{3}	0.00	0.00
Np-239	9.66	2.24×10^{6}	4.73×10^{-6}	1.10
Np-240	1.62×10^{-4}	1.95×10^{3}	0.00	0.00
Pd-107	2.85×10^{1}	1.47×10^{-2}	2.84×10^{1}	1.46×10^{-2}
Pd-109	3.86×10^{-2}	8.27×10^{4}	0.00	0.00
Pm-146	1.47×10^{-3}	6.51×10^{-1}	1.36×10^{-3}	6.02×10^{-1}
Pm-147	2.68×10^{1}	2.49×10^{4}	2.39×10^{1}	2.22×10^{4}
Pr-143	3.03	2.04×10^{5}	4.55×10^{-5}	3.06
Pr-144	1.59×10^{-3}	1.20×10^{5}	9.25×10^{-4}	6.99×10^{4}
Pr-144m	0.00	0.00	3.68×10^{-6}	6.67×10^{2}
Pr-145	5.01×10^{-2}	1.81×10^{5}	0.00	0.00
Pu-237	6.55×10^{-5}	7.97×10^{-1}	2.32×10^{-6}	2.82×10^{-2}
Pu-238	8.69	1.49×10^{2}	9.13	1.56×10^{2}
Pu-239	5.55×10^{3}	3.45×10^{2}	5.57×10^{3}	3.46×10^{2}
Pu-240	2.45×10^{3}	5.59×10^{2}	2.45×10^{3}	5.59×10^{2}
Pu-241	3.16×10^{2}	3.26×10^{4}	3.06×10^{2}	3.15×10^{4}
Pu-242	1.36×10^{2}	5.34×10^{-1}	1.36×10^{2}	5.34×10^{-1}
Pu-243	5.58×10^{-3}	1.45×10^{4}	0.00	0.00
Pu-244	3.43×10^{-4}	6.09×10^{-9}	3.43×10^{-4}	6.09×10^{-9}
Pu-245	1.72×10^{-8}	2.10×10^{-2}	0.00	0.00
Pu-246	3.02×10^{-11}	1.48×10^{-6}	0.00	0.00
Rb-86	8.44×10^{-3}	6.91×10^{2}	2.35×10^{-6}	1.92×10^{-1}
Rb-87	1.24×10^{1}	1.08×10^{-6}	1.24×10^{1}	1.08×10^{-6}
Rb-88	8.31×10^{-4}	9.97×10^{4}	0.00	0.00
Rb-89	9.14×10^{-4}	1.08×10^{5}	0.00	0.00
Rh-105	2.96×10^{-1}	2.50×10^{5}	0.00	0.00
Rh-106	2.78×10^{-5}	9.90×10^{4}	1.84×10^{-5}	6.55×10^{4}
Rh-107	1.87×10^{-3}	1.52×10^{5}	0.00	0.00
Ru-105	3.82×10^{-2}	2.57×10^{5}	0.00	0.00
Ru-106	2.98×10^{1}	9.97×10^{4}	1.97×10^{1}	6.59×10^{4}
Sb-125	2.40	2.48×10^{3}	2.07	2.14×10^{3}
Sb-126	6.31×10^{-3}	5.27×10^{2}	2.74×10^{-8}	2.29×10^{-3}
Sb-126m	0.00	0.00	0.00	0.00
Sb-127	1.15×10^{-1}	3.07×10^{4}	0.00	0.00
Sb-129	1.38×10^{-2}	7.76×10^{4}	0.00	0.00
Sb-130	6.51×10^{-4}	2.35×10^{4}	0.00	0.00
Se-79	4.01×10^{-1}	2.79×10^{-2}	4.01×10^{-1}	2.79×10^{-2}
Sm-147	3.90	8.95×10^{-8}	0.00	0.00
Sm-151	1.09×10^{1}	2.87×10^{2}	0.00	0.00
Sm-153	5.50×10^{-2}	2.41×10^{4}	0.00	0.00

Isotope	Single 6% Burnup Assembly mass (grams)	Single 6% Burnup Assembly (curies)	VTR Assembly Decayed 220 Days mass (grams)	VTR Assembly Decayed 220 Days (curies)
Sr-89	3.56	1.03×10^{5}	1.74×10^{-1}	5.05×10^{3}
Sr-90	2.47×10^{1}	3.37×10^{3}	2.43×10^{1}	3.32×10^{3}
Sr-91	4.67×10^{-2}	1.69×10^{5}	0.00	0.00
Sr-92	1.54×10^{-2}	8.90×10^{6}	0.00	0.00
Tc-101	2.73×10^{-3}	3.58×10^{5}	0.00	0.00
Tc-104	3.30×10^{-3}	3.28×10^{5}	0.00	0.00
Tc-99	6.03×10^{1}	1.02	6.10×10^{1}	1.03
Tc-99m	0.00	0.00	0.00	0.00
Te-125m	0.00	0.00	2.65×10^{-2}	4.77×10^{2}
Te-127	1.12×10^{-2}	2.96×10^{4}	1.70×10^{-5}	4.49×10^{1}
Te-127m	0.00	0.00	4.85×10^{-3}	4.57×10^{1}
Te-129	3.47×10^{-3}	7.27×10^{4}	2.04×10^{-8}	4.27×10^{-1}
Te-129m	1.43×10^{1}	0.00	2.25×10^{-5}	6.78×10^{-1}
Te-131	3.32×10^{-3}	1.91×10^{5}	0.00	0.00
Te-131m	0.00	0.00	0.00	0.00
Te-132	9.37×10^{-1}	2.84×10^{5}	0.00	0.00
Te-133	1.99×10^{-3}	2.26×10^{5}	0.00	0.00
Te-133m	0.00	0.00	0.00	0.00
Te-134	8.85×10^{-3}	2.97×10^{5}	0.00	0.00
U-232	9.03×10^{-6}	1.93×10^{-4}	2.38×10^{-5}	5.10×10^{-4}
U-233	1.13×10^{-5}	1.09×10^{-7}	1.31×10^{-5}	1.27×10^{-7}
U-234	4.18×10^{-1}	2.61×10^{-3}	4.61×10^{-1}	2.88×10^{-3}
U-235	1.20×10^{3}	2.60×10^{-3}	1.20×10^{3}	2.60×10^{-3}
U-236	7.93×10^{1}	5.13×10^{-3}	7.94×10^{1}	5.14×10^{-3}
U-237	3.82×10^{-1}	3.12×10^{4}	9.54×10^{-6}	7.79×10^{-1}
U-238	2.98×10^{4}	1.00×10^{-2}	2.97×10^{4}	9.99×10^{-3}
Xe-131m	1.43×10^{1}	0.00	1.32×10^{-7}	1.11×10^{-2}
Xe-133	0.00	3.41×10^{5}	5.20×10^{-13}	9.73×10^{-8}
Xe-133m	0.00	0.00	0.00	0.00
Xe-135	0.00	4.01×10^{5}	0.00	0.00
Xe-135m	0.00	0.00	0.00	0.00
Xe-138	2.80×10^{-3}	2.67×10^{5}	0.00	0.00
Y-90	1.43×10^{1}	3.62×10^{3}	6.18×10^{-3}	3.36×10^{3}
Y-91	5.60	1.37×10^{5}	4.17×10^{-1}	1.02×10^{4}
Y-91m	0.00	0.00	0.00	0.00
Y-92	2.03×10-2	1.95×10^{5}	0.00	0.00
Y-93	7.13×10-2	2.38×10^{5}	0.00	0.00
Y-94	2.32×10^{-3}	2.43×10^{5}	0.00	0.00
Y-95	1.45×10^{-3}	2.68×10^{5}	0.00	0.00
Zr-93	4.13×10^{1}	1.04×10-1	4.15×10^{1}	1.04×10^{-1}
Zr-95	1.06×10^{1}	2.28×10^{5}	9.89×10-1	2.12×10^{4}
Zr-97	1.51×10-1	1.20×10^{4}	0.00	0.00
Total per Assembly	**4.02×10^{4}**	**2.35×10^{7}**	**3.99×10^{4}**	**4.30×10^{5}**

Note: These are the radiologically significant 143 isotopes of the 768 isotope values provided.
Source: ECAR-4777 (INL 2020b), CSDR December 2019 (INL 2020c), or Grabaskas 2019 (ECAR-4737) for single 6 percent burnup assembly mass; email from Jason Andrus (BEA 2020) for 220 day decayed spent fuel assembly; Federal Guidance Report 13 for isotope half-life values.

D.8 Attachment D2: Isotopic Composition of 4-Year Cooled VTR Fuel

The following table presents the isotopic composition of a 4-year cooled VTR fuel assembly.

Table D–43. Isotopic Composition of a 4-Year Cooled VTR Assembly

Isotope	VTR Assembly Decayed 4 Years (curies)	Isotope	VTR Assembly Decayed 4 Years (curies)	Isotope	VTR Assembly Decayed 4 Years (curies)
Ac-225	6.65×10^{-11}	La-138	3.58×10^{-11}	Pu-243	1.45×10^{-10}
Ac-227	1.34×10^{-8}	Nb-93m	1.61×10^{-2}	Pu-244	6.20×10^{-9}
Am-241	$2.38\times10^{+2}$	Nb-94	3.02×10^{-5}	Pu-246	2.74×10^{-20}
Am-243	$1.10\times10^{+0}$	Nb-95	6.94×10^{-2}	Ra-222	1.33×10^{-29}
Am-245	7.01×10^{-17}	Nb-95m	3.60×10^{-4}	Ra-223	1.34×10^{-8}
Am-246m	2.74×10^{-20}	Nd-144	6.42×10^{-11}	Ra-224	9.40×10^{-4}
At-217	6.65×10^{-11}	Np-237	8.13×10^{-3}	Ra-225	6.65×10^{-11}
Ba-137m	$6.62\times10^{+3}$	Np-239	$1.10\times10^{+0}$	Ra-226	1.02×10^{-10}
Bi-210	4.47×10^{-11}	Np-240	7.42×10^{-12}	Rb-86	1.77×10^{-21}
Bi-211	1.34×10^{-8}	Np-240m	6.19×10^{-9}	Rb-87	1.06×10^{-6}
Bi-212	9.40×10^{-4}	Pa-231	2.19×10^{-7}	Rh-102	1.65×10^{-2}
Bi-213	6.65×10^{-11}	Pa-233	8.13×10^{-3}	Rh-103m	1.75×10^{-6}
Bi-214	1.02×10^{-10}	Pa-234	1.60×10^{-5}	Rh-106	$6.48\times10^{+3}$
Bk-249	4.83×10^{-12}	Pa-234m	9.98×10^{-3}	Rn-218	2.03×10^{-17}
Cd-113m	1.90×10^{-5}	Pb-209	6.65×10^{-11}	Rn-219	1.34×10^{-8}
Cd-115m	5.19×10^{-13}	Pb-210	4.47×10^{-11}	Rn-220	9.40×10^{-4}
Ce-139	2.51×10^{-3}	Pb-211	1.34×10^{-8}	Rn-222	1.02×10^{-10}
Ce-141	7.43×10^{-9}	Pb-212	9.40×10^{-4}	Ru-103	1.77×10^{-6}
Ce-144	$3.42\times10^{+3}$	Pb-214	1.02×10^{-10}	Ru-106	$6.48\times10^{+3}$
Cm-241	7.90×10^{-17}	Pd-107	1.46×10^{-2}	Sb-124	7.79×10^{-6}
Cm-242	$5.12\times10^{+0}$	Pm-146	3.95×10^{-1}	Sb-125	$9.16\times10^{+2}$
Cm-243	2.45×10^{-1}	Pm-147	$9.03\times10^{+3}$	Se-79	6.16×10^{-3}
Cm-244	$2.90\times10^{+1}$	Po-210	4.38×10^{-11}	Sm-146	2.16×10^{-8}
Cm-245	1.68×10^{-3}	Po-211	3.69×10^{-11}	Sm-147	5.09×10^{-7}
Cm-246	4.55×10^{-5}	Po-212	6.02×10^{-4}	Sm-151	$2.80\times10^{+2}$
Cm-247	1.45×10^{-10}	Po-213	6.50×10^{-11}	Sn-119m	1.63×10^{-7}
Cm-248	4.51×10^{-11}	Po-214	1.02×10^{-10}	Sn-121	4.70×10^{-9}
Cs-134	$5.75\times10^{+2}$	Po-215	1.34×10^{-8}	Sn-121m	6.06×10^{-9}
Cs-135	1.15×10^{-1}	Po-216	9.40×10^{-4}	Sn-123	2.15×10^{-12}
Cs-137	$6.99\times10^{+3}$	Po-218	1.02×10^{-10}	Sr-89	2.08×10^{-4}
Eu-152	7.88×10^{-1}	Pr-144	$3.42\times10^{+3}$	Sr-90	$3.10\times10^{+3}$
Eu-154	$1.43\times10^{+2}$	Pr-144m	$3.27\times10^{+1}$	Tc-98	6.90×10^{-7}
Eu-155	$6.83\times10^{+2}$	Pu-236	2.23×10^{-2}	Tc-99	$1.04\times10^{+0}$
Fr-221	6.65×10^{-11}	Pu-237	1.85×10^{-10}	Te-125m	$2.24\times10^{+2}$
Fr-223	1.85×10^{-10}	Pu-238	$1.57\times10^{+2}$	Te-127	1.69×10^{-2}
I-129	3.22×10^{-3}	Pu-239	$3.45\times10^{+2}$	Te-127m	1.72×10^{-2}
In-115m	5.52×10^{-17}	Pu-240	$5.56\times10^{+2}$	Te-129	3.32×10^{-12}
Kr-81	1.76×10^{-8}	Pu-241	$2.69\times10^{+04}$	Te-129m	5.27×10^{-12}
Kr-85	3.35×10^{-3}	Pu-242	5.38×10^{-1}	Th-226	1.33×10^{-29}
Th-227	1.32×10^{-8}				
Th-228	9.40×10^{-4}				
U-238	9.98×10^{-3}				
U-240	6.19×10^{-9}				

Isotope	*VTR Assembly Decayed 4 Years (curies)*	*Isotope*	*VTR Assembly Decayed 4 Years (curies)*	*Isotope*	*VTR Assembly Decayed 4 Years (curies)*
Xe-127	2.19×10^{-14}				
Y-89m	2.00×10^{-8}				
Y-90	$3.10\times10^{+3}$				
Y-91	4.27×10^{-3}				
Zr-93	1.04×10^{-1}				
Zr-95	3.15×10^{-2}				

VTR = Versatile Test Reactor.
Source: ECAR No. 5093, *VTR Assembly Source Term at Four-Years Decay Time*, TEM-326, Rev. 1, 01/07/2020.

D.9 Attachment D3: Nuclides Released from Inadvertent Nuclear Criticality

Table D–44. Curies of Important Nuclides Released During Nuclear Excursion Involving Plutonium Solution

Nuclide	Half-life	Radioactivity, Ci [a]		
		0 to 0.5 hour	0.5 to 8 hour	Total
Kr-83m	1.8 h	1.5×10^{1}	9.5×10^{1}	1.1×10^{2}
Kr-85m	4.5 yr	9.4	6.1×10^{1}	7.1×10^{1}
Kr-85	1.7 yr	1.2×10^{-4}	7.2×10^{-4}	8.1×10^{-3}
Kr-87	76.3 m	6.0×10^{1}	3.7×10^{2}	4.3×10^{2}
Kr-88	2.8 h	3.2×10^{1}	2.0×10^{2}	2.3×10^{2}
Kr-89	3.2 m	1.8×10^{3}	1.1×10^{4}	1.3×10^{4}
Xe-131m	11.9 d	1.4×10^{-2}	8.6×10^{-2}	1.0×10^{-1}
Xe-133m	2.0 d	3.1×10^{-1}	1.9	2.2
Xe-133	5.2 d	3.8	2.3×10^{1}	2.7×10^{1}
Xe-135m	15.6 m	4.6×10^{2}	2.8×10^{3}	3.3×10^{3}
Xe-135	9.1 h	5.7×10^{1}	3.5×10^{2}	4.1×10^{2}
Xe-137	3.8 m	6.9×10^{3}	4.2×10^{4}	4.9×10^{4}
Xe-138	14.2 m	1.5×10^{3}	9.5×10^{3}	1.1×10^{4}
I-131	8.0 d	1.5	9.5	1.1×10^{1}
I-132	2.3 h	1.7×10^{2}	1.0×10^{3}	1.2×10^{3}
I-133	20.8 h	2.2×10^{1}	1.4×10^{2}	1.6×10^{2}
I-134	52.6 m	6.0×10^{2}	3.7×10^{3}	4.3×10^{3}
I-135	6.6 h	6.3×10^{1}	3.9×10^{2}	4.5×10^{2}
Pu-238 [b]				5.9×10^{-4}
Pu-239 [b]				2.7×10^{-5}
Pu-240 [b]				5.8×10^{-5}
Pu-241 [b]				1.8×10^{-2}
Pu-242 [b]				4.3×10^{-7}
Am-241 [b]				2.4×10^{-5}

Am = americium; Ci = curies; d = day; h = hour; I = iodine; Kr = krypton; m = minute; Pu = plutonium; Xe = xenon; y = year.

[a] Total Ci, except for Pu and Am, are based on cumulative yield for fission energy spectrum. The assumption of cumulative yield is conservative, i.e., it does not consider appropriate decay schemes. Calculations regarding individual nuclide yields and decay schemes may be considered on an individual basis. Data in this table do not include the iodine reduction factor.

[b] Total radioactivity assumes the isotopic mix to be the equilibrium mix for recycled plutonium and 1 milligram of plutonium oxide released.

Source: DOE 1994.